RAIO-0119-64081

January 09, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 465 (eRAI No. 9494) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 465 (eRAI No. 9494)," dated May 04, 2018
2. NuScale Technical Report Containment Response Analysis Methodology, dated January 2017, TR-0516-49084

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosures to this letter contain NuScale's response to the following RAI Question from NRC eRAI No. 9494:

- 06.02.01.01.A-17

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 465 (eRAI No. 9494). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter and the enclosed responses make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Paul Infanger at 541-452-7351 or at pinfanger@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9494, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 9494, nonproprietary

Enclosure 3: Affidavit of Zackary W. Rad, AF-0119-64082

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9494, proprietary

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Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 9494, nonproprietary

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9494
Date of RAI Issue: 05/04/2018

NRC Question No.: 06.02.01.01.A-17

The "Containment Response Analysis Methodology" Technical Report (CRAM TeR) (TR-0516-49084-P Rev. 0), in Section 2.0, states that the qualification of the LOCA and non-LOCA methodologies presented in both LOCA and Non-LOCA topical reports (TR-0516-49422- P, Rev. 0, and TR-0516-49416-P, Rev. 0, respectively) and in particular the comparisons to separate effects tests and integral effects tests, are applicable for the containment response analysis methodology. It appears to the staff that NuScale is relying on the qualification documented in these two topical reports as part of its containment response analysis methodology (CRAM). However, the LOCA topical report (TR) states that "NuScale is requesting Nuclear Regulatory Commission (NRC) review and approval to use the LOCA evaluation model (EM) described in this report for analyses of *design-basis LOCA events* in the NPM." Therefore, the staff is concerned about the applicability of the NIST-1 separate/integral effects tests results, validation, and distortion analysis presented in the LOCA TR, to the NRELAP5 safety analysis models used in the CRAM TeR to evaluate containment peak pressure/temperature for the design-basis events. The staff has the same concern about the stated scope for the non-LOCA TR and its applicability to the CRAM. Likewise, the scaling distortion report (Calculations to Support NIST-1 Distortion Analysis and Modeling of Containment and Pool heat Transfer, {{ }}2(a),(c)) that is used to analyze the scaling distortions of the NIST-1 testing for both LOCA EM and CRAM, is only discussed in the LOCA TR. Sections 4.1.1 and 4.1.2 of the CRAM TeR state that no additional qualification activities were performed for the LOCA and non-LOCA models relative to applicability to the CRAM, and that these qualification activities were adequate. However, a clear and complete basis for that conclusion was not provided. NuScale is therefore requested to address the following questions, and update the FSAR and the reports involved, accordingly:

a. Clarify the intended applicability of the LOCA and non-LOCA TRs to the CRAM, including specification of the portions of those two TRs considered applicable to the CRAM,

b. Make any necessary associated changes to the scope of the LOCA TR and the non-LOCA TR, and

c. Demonstrate the applicability of those portions of the LOCA and non-LOCA TRs to the CRAM, and justify that the qualification activities in those TRs were adequate.

NuScale Response:

Part (a):

In the containment response analysis methodology described in TR-0516-49084-P, the qualification of the NRELAP5 code to predict the NuScale Power Module (NPM) containment response is based on the code qualification and the NPM plant modeling approach that is described in the LOCA evaluation model topical report TR-0516-49422-P (for primary side pipe breaks and reactor valve opening events), and in the non-LOCA evaluation model topical report TR-0516-49416-P (for main steam line and feedwater line break events). As identified in the markup of TR-0516-49084-P included in this response, the containment response analysis methodology is an extension of the NuScale LOCA, inadvertent valve opening, and non-LOCA methodologies that were developed following the guidance of Regulatory Guide 1.203. However, the containment response analysis methodology was not independently developed following the guidance of RG 1.203. The containment response analysis report references these methodologies and identifies and justifies differences for the containment response analysis when compared to these methodologies.

The containment response analysis methodology report describes changes to the NPM plant modeling that are necessary to conservatively bias the model to maximize mass and energy releases and maximize the containment pressure and temperature response. For the NPM design, changes to the NPM plant model are required because NPM plant calculations done in accordance with the LOCA EM described in TR-0516-49422-P are conservatively biased to demonstrate that the NPM maintains margin to the acceptance criteria of level above the top of the core, and to demonstrate that CHF does not occur. Similarly, calculations done with the NRELAP5 code in accordance with the non-LOCA EM described in TR-0516-49416-P are conservatively biased to demonstrate that the NPM maintains margin to the acceptance criteria of maximum primary and secondary pressure, to provide boundary conditions for downstream subchannel analysis and radiological analysis, and demonstrate that event escalation does not occur. These evaluation models are developed to demonstrate margin to these specific acceptance criteria.

Although demonstrating margin to containment pressure/temperature acceptance criteria are not the purpose of, or within scope of, the calculations done in accordance with the LOCA or non-LOCA evaluation models, it is necessary for these evaluation models to adequately predict the containment response in order to demonstrate that LOCA acceptance criteria are met (or non-LOCA acceptance criteria for a feedwater line break or main steam line break event), due to the integrated nature of the NPM design. Therefore, these evaluation models provide an appropriate initial model or starting point for which changes made in accordance with the NPM containment pressure/temperature analysis methodology maximize containment pressure/temperature response. Specific aspects of the EMs that demonstrate the code and model applicability for the containment response analysis are discussed in part (c) of this response.

Part (b):

Calculations performed in accordance with the LOCA or non-LOCA EM are not used to demonstrate margin to the containment pressure and temperature acceptance criteria. Therefore no changes to the scope of the LOCA or non-LOCA EMs are required. As discussed in response to part (a) and part (c), these evaluation models are developed to demonstrate margin to the acceptance criteria identified therein. Due to the integrated nature of the NPM design these EMs provide an appropriate starting point for the NPM containment pressure/temperature analysis methodology. The initial and boundary conditions are biased to produce conservative containment pressure and temperature response in accordance with the containment response analysis methodology.

The following discussion provides additional clarification for part of the scope statement in the LOCA EM .

Section 1.2 of the LOCA EM states:

> *'Application of the EM demonstrates that fuel does not experience CHF conditions, collapsed water level remains above the top of the active fuel, and containment remains intact and pressure and temperature remain below design limits. This assures that no fuel failure occurs and the acceptance criteria of the 10 CFR 50.46 (Reference 3), excluding long-term cooling, are satisfied.'*

It is further clarified in Section 4.3 of the LOCA EM that:

> *'To ensure ECCS performance, the containment must be intact and remain below pressure and temperature design limits. Consequently, peak containment pressure and temperature are evaluated to ensure compliance with 50.46 criteria. However, the peak containment pressure and temperature for containment performance are calculated with a different methodology.'*

In the context of calculations performed in accordance with the LOCA EM, it is necessary to check that containment pressure and temperature design limits are not exceeded in the calculations to ensure ECCS performance and compliance with 50.46 criteria. However, for the purpose of demonstrating margin to the containment pressure and temperature acceptance criteria, calculations are performed as described in the containment response analysis technical report.

Part (c):

The following response discusses portions of the LOCA and non-LOCA TRs applicable to the containment response analysis methodology to justify that the qualification activities in these topical reports were adequate for the purpose of containment response analysis.

LOCA Pipe Breaks

As part of the LOCA EM development, the adequacy of the NRELAP5 code to predict the high ranked phenomena associated with a LOCA or pipe break inside containment was evaluated. Therefore, the LOCA EM provides an appropriate starting point for evaluation of the NPM peak containment pressure and temperature response. Key parts of the LOCA EM demonstrating that NPM containment pressure response was considered over the course of the development of the LOCA EM are summarized below.

As specified in the LOCA topical report TR-0516-49422-P, Section 1.2, the LOCA EM is applicable for:

> *The EM is applicable to a nuclear power plant that follows the general description of the NuScale Power Plant design in Section 3.0. Applicability of the EM is based on the NuScale LOCA PIRT, which identifies and ranks those phenomena the EM must be qualified to model during a LOCA in an NPM.*

The LOCA topical report Section 4 describes the development of the NPM LOCA PIRT for postulated breaks in the reactor coolant pressure boundary that result in leakage of reactor coolant at a rate exceeding the capability of the normal reactor coolant makeup system. As described in Section 4.3 of the LOCA topical report, peak containment pressure and temperature were considered figures of merit in the development of the LOCA PIRT because the containment must be intact and remain below pressure and temperature design limits to ensure ECCS performance. Table 4-4 summarizes high ranked phenomena identified in the LOCA PIRT development, including the following phenomena identified as important for the prediction of the containment response:

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}} 2(a),(c)

Section 5.0 of the LOCA topical report describes the NPM LOCA model. The NPM LOCA model is consistent with the SET and IET assessment used to validate NRELAP5. Section 5.1.1 of the LOCA topical report describes the general model nodalization. As described in Section 3.2.4.1 of the containment response analysis technical report, the NPM geometry inputs and conservative fuel inputs in the containment response analysis model are consistent with those used by the LOCA EM.

In the LOCA topical report, Section 5.2 through Section 5.5 describe the NPM analysis setpoints and trips, initial plant conditions, LOCA break spectrum, and sensitivity studies performed for calculations done in accordance with the LOCA EM. For calculations to assess the containment response analysis methodology, the NPM setpoints and trips, initial plant condition biasing, break spectrum and sensitivities performed are as described in the containment response analysis technical report in Section 3.5.1 and Section 3.5.2.

Section 6 of the LOCA topical report describes the NRELAP5 code. The NRELAP5 code as described in Section 6 of the LOCA topical report is used in the containment response analysis calculations.

Section 7 of the LOCA topical report summarizes the results of NRELAP5 validation against experimental data. Assessment against the NIST-1 integral effects tests HP-06, HP-06b, HP-07, HP-09 included examination of the peak CNV pressure response as part of the overall assessment of the CNV pressure response prediction.

Section 8 of the LOCA topical report summarizes the adequacy of the NRELAP5 code for analysis of design-basis LOCAs. This is based on:

- Bottom-up assessment of the NRELAP5 models and correlations to determine their adequacy to predict the high ranked phenomena identified in the PIRT.
- Top-down assessment of the LOCA EM including review of the EM governing equations and numerics to determine their applicability to NPM LOCA analysis, and evaluation of the integral code performance based on the assessments of the EM against relevant integral effects tests.

Section 8.4 of the LOCA topical report summarizes the adequacy findings. From the bottom-up evaluation, {{

}}2(a),(c)

In the LOCA EM, Table 8-14 summarizes the high-ranked phenomena from the LOCA PIRT and the findings from the top-down adequacy evaluation based on the NIST-1 integral effects tests.

As summarized in the LOCA topical report Section 8.4.3, overall, the LOCA EM demonstrated that the NRELAP5 code is applicable to predict the high-ranked phenomena that govern the LOCA response in the NPM. The NPM containment response was identified as a figure of merit in the LOCA PIRT and considered through the LOCA EM development as summarized in this response. Therefore these aspects of the LOCA EM are applicable for the containment response analysis and the NRELAP5 code qualification demonstrated as part of the LOCA EM development is adequate for the containment response analysis using the NPM code and no additional qualification activities are necessary.

Inadvertent Opening of RPV Valves

In the LOCA topical report TR-0516-49422-P, Appendix B describes the evaluation model and methodology applied by NuScale to analyze inadvertent opening of reactor pressure vessel valves (IORV). The methodology and evaluation model were developed by extending the LOCA methodology presented in the main body of TR-0516-49422-P. As summarized in Appendix B.7, the NRELAP5 assessments discussed in Section 7 in support of the LOCA EM are also applicable for the IORV EM because the physical phenomena and their importance ranking between the two scenarios are the same. Consequently, the bottom-up and top-down applicability evaluations presented in Section 8.2 and 8.3 are also valid for the IORV EM. Therefore, the discussion in this response for containment response of pipe break LOCAs is also applicable for containment response to inadvertent valve opening events.

Appendix B.7 of the LOCA topical report also provides additional NRELAP5 assessment results for NIST-1 test HP-43, an updated RVV spurious opening test.

In addition, the NRELAP5 code was assessed against the NIST-1 HP-49 test, which is an integral effects test modeling a spurious RRV opening into containment. The response to RAI 9459, Question 06.02.01.01.A-21 provided a summary of the test and a summary of the test assessment was included as Appendix C in the LOCA EM topical report TR-0516-49422.

<u>Non-LOCA Main Steam Line and Feedwater Line Breaks</u>

In the non-LOCA EM described in topical report TR-0516-49416-P, the NRELAP5 code is used to calculate the NPM system thermal-hydraulic response to non-LOCA events, including the main steam line break and feedwater line break events that are the secondary side break events analyzed for the containment response. In the non-LOCA EM, the use of the NRELAP5 code to predict applicable high-ranked phenomena associated with the non-LOCA transients and accidents was evaluated. Therefore, the non-LOCA EM provides an appropriate starting point for evaluation of the NPM peak containment pressure and temperature response for secondary side release events. Key parts of the non-LOCA EM demonstrating that the NPM containment pressure response was considered over the course of the development of the LOCA EM are summarized below.

As specified in the non-LOCA topical report TR-0516-49416-P Section 1.2:

- *The non-LOCA evaluation model uses the NRELAP5 code to perform system transient analysis of the NPM design basis events listed in Table 4-1… The NRELAP5 code is described in the LOCA evaluation model …*
- *The non-LOCA evaluation model is applicable to a nuclear power plant that follows the general description of the NuScale plant design in Section 3.0. The applicability of the EM is based on the non-LOCA phenomena identification and ranking table and*

assessment of the high-ranked phenomena that are treated as part of the system transient analysis.

Section 5.1 of the non-LOCA topical report describes the development of the non-LOCA PIRT and evaluation of high-ranked phenomena. In development of the PIRT, the expert panel evaluated five design-basis non-LOCA events representing different categories of non-LOCA events. The events included a main steam line break representing events that result in an increase in heat removal from the RCS and a feedwater line break representing events that result in a decrease in heat removal from the RCS. In the PIRT, the containment pressure was identified as a figure of merit in phase 2 of the non-LOCA transient (post-reactor trip transition to stable DHRS cooling). High ranked phenomena associated with CNV pressure response include:

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}}2(a),(c)

Section 6 of the non-LOCA topical report describes the NuScale non-LOCA plant transient model.

As described in the containment response technical report TR-0516-49084 Section 3.2.4.2, the NRELAP5 model used for secondary system pipe break analysis in the containment response analysis methodology is similar to the NRELAP5 model used in the non-LOCA accident FSAR Chapter 15 analyses. The model for the secondary side pipe break containment analysis conservatively maximizes the mass and energy release and minimizes containment heat removal. For calculations to assess the containment response analysis methodology, the NPM setpoints and trips, initial plant condition biasing, break spectrum and sensitivities performed are as described in the containment response analysis technical report in Section 3.5.3, Section 3.5.4, Section 3.5.5, and Section 3.5.6. As summarized in the containment response analysis technical report Section 4.1.2, the body of NIST-1 separate effects and LOCA integral tests have demonstrated the capability of NRELAP5 to adequately model the NPM design. High ranked phenomena associated with secondary side line breaks and containment pressure response were considered as part of the non-LOCA EM development. In addition, the secondary system mass and energy release analyses for the NPM are non-limiting compared to the primary system containment response analysis. Therefore, the NRELAP5 qualification for non-LOCA events is adequate to demonstrate qualification of the code for containment response analysis and no additional qualification activities are necessary.

Updates to TR-0516-49084 to reference LOCA EM Appendix B for the qualification the valve opening event models and their applicability for containment response analysis are attached to this response. The update includes a discussion of LOCA EM, Appendix B.7 that provides assessment results for a spurious RVV opening event test, along with adding a reference to LOCA EM Appendix C for assessment of the NIST-1 spurious RRV opening event test. The markups provided with this response do not update the CNV design pressure.

Impact on DCA:

Technical Report TR-0516-49084, Containment Response Analysis Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

Executive Summary

This report presents the NuScale Power, LLC, (NuScale) methodology used to analyze the mass and energy release into the containment vessel (CNV) for the spectrum of design basis transients and accidents, and the resulting pressure and temperature response of the CNV. The NuScale Power Module (NPM) limiting peak pressure and temperature results determined using the methodology are presented.

The containment response analysis methodology uses the NRELAP5 thermal-hydraulic code, which is a NuScale-modified version of the RELAP5-3D© v 4.1.3 code used for loss-of-coolant accident (LOCA) and non-LOCA transient and accident analyses, including the response of the CNV.

The NRELAP5 model used to model NPM performance for primary system LOCA and emergency core cooling system valve-opening event analyses is similar ~~with~~ to the model used in the LOCA evaluation model, described by Reference 7.2.1. The NRELAP5 model used for secondary system pipe-break analysis in the containment response analysis methodology is ~~consistent with~~ similar to the non-LOCA model described by the Non-LOCA Evaluation Model Report (Ref: 7.2.2). Changes made to these models that maximize containment pressure and temperature response to primary and secondary system release events are described in this report. These changes conservatively maximize the mass and energy release and minimize the performance of the containment heat removal system and are consistent with acceptance criteria given by Design Specific Review Standard Section 6.2.1.3 (Ref: 7.1.6) and Design Specific Review Standard Section 6.2.1.4 (Ref: 7.1.7).

~~Other differences exist between the NRELAP5 model used to model NPM performance for primary system LOCA and emergency core cooling system valve-opening event analyses and the containment analysis model. These modeling differences, identified in Section 3.2.4.1, have a negligible impact on the CNV analysis results.~~

Initial and boundary conditions for the spectrum of primary system release containment response analyses and secondary system pipe break analyses are selected to ensure a conservative CNV peak pressure and peak temperature result. These initial and boundary conditions are described in this report, along with the rationale for their selection.

The results of the NRELAP5 limiting analyses using the containment response analysis methodology are presented in this report. These analyses cover the spectrum of primary system mass and energy release scenarios for the NPM, and secondary system pipe break scenarios.

The limiting LOCA peak pressure and CNV wall temperature are a result of the reactor coolant system (RCS) injection line break. The LOCA limiting peak CNV wall temperature is approximately ~~523~~526 degrees F and it results from a reactor coolant system injection line break case, with a loss of normal alternating current (AC) power. The LOCA limiting peak internal pressure is approximately ~~921~~959 psia, which ~~also~~ results from a reactor coolant system injection line break case with a loss of normal AC and DC power. The LOCA event peak CNV pressure is below the CNV design pressure of ~~1000~~1075 psia. The LOCA peak CNV pressure and wall temperature bound the main steamline break (MSLB) and feedwater line break (FWLB) results.

2.0 Background

The CNV is a compact, steel pressure vessel that consists of an upright cylinder with top and bottom head closures. The CNV is partially immersed in a below-grade reactor pool that provides a passive heat sink and is absent of internal sumps or subcompartments that could entrap water or gases. The CNV and the reactor pool are housed within a Seismic Category 1 Reactor Building. The unique nature of the NPM design necessitates development of a specific containment response analysis methodology.

This technical report describes the thermal-hydraulic accident analysis methodology for primary and secondary system M&E releases into the CNV of the NPM, and the resulting pressure and temperature response of the CNV. This report presents the bases for the analysis methodology and results in support of Chapter 6 of the NuScale Final Safety Analysis Report (FSAR). The containment response analysis methodology and CNV peak pressure and temperature results are compared to applicable regulatory guidance, including the Design Specific Review Standard for NuScale Small Modular Reactor (SMR) Design, Section 6.2.1 (Ref: 7.1.4). A spectrum of M&E release events is analyzed that bounds all of the LOCAs and valve-opening transients in the primary system and all secondary-system pipe-break accidents. The containment response analysis methodology uses conservative initial conditions and boundary conditions to ensure overall conservative results. The limiting results are shown to be less than the design pressure (~~1000~~1075 psia) and the design temperature (550 degrees F) of the CNV.

The qualification of the LOCA, valve opening event and non-LOCA methodologies presented in Reference~~s~~ 7.2.1 and Reference 7.2.2, in particular the comparisons to separate effects tests and integral effects tests, are applicable for the containment response analysis methodology presented in this report. The differences in the NRELAP5 simulation models used in the containment response analysis methodology as compared to the LOCA, valve opening event and non-LOCA models, along with the rationale for the selection of conservative initial and boundary conditions, are the subject of this report. Analysis results are presented for the limiting cases~~, along with nominal condition case results, demonstrating conservatism in certain initial conditions~~.

2.1 Regulatory Requirements

The Nuclear Regulatory Commission (NRC) regulations and regulatory guidance applicable to the containment response analysis methodology are described in this section. The elements of the containment response analysis methodology that address each of these regulations and requirements are discussed.

2.1.1 10 CFR 50 Appendix A - General Design Criteria for Nuclear Power Plants

The General Design Criteria (GDC) for Nuclear Power Plants, Appendix A to 10 CFR 50 (Ref: 7.1.2), include the NRC regulations applicable to the containment response methodology. Compliance with GDC 16 and 50 and PDC 38 is as follows:

General Design Criterion 16 - The analyses performed per the containment response analysis methodology are used to establish the limiting CNV pressure and temperature conditions resulting from the spectrum of design-basis primary system and secondary system M&E releases resulting from pipe breaks and valve actuations. The CNV is designed to ensure that the design pressure and temperature limit are not exceeded as demonstrated by the analysis results.

Principal Design Criterion 38 - The analyses performed per the containment response analysis methodology establish the performance of NPM containment heat removal and demonstrate that the containment peak pressure and temperature are rapidly reduced. The methodology addresses LOCAs, valve-opening events and secondary pipe breaks. Following containment isolation and opening of the ECCS valves, the containment heat removal function is passive and does not require electric power. The requirement to rapidly reduce the containment pressure and temperature is demonstrated by the peak pressure decreasing to less than 50 percent of the peak value consistent with Design Specific Review Standard (DSRS) Section 6.2.1.1.A (Ref: 7.1.5). Potential single failures have been considered in the methodology, and the results of the analyses show that the safety functions can be performed including the limiting single failure.

General Design Criterion 50 - The analyses performed per the containment response analysis methodology demonstrate that sufficient margin to the CNV design pressure and temperature is maintained. The methodology explicitly models all energy sources including energy in the steam generators (SGs). However, the energy from the post-LOCA oxidation of the cladding that is typical of light water reactors is not applicable to the NuScale design and is not included. Calculated cladding temperatures for design basis LOCAs are below the level where cladding oxidation occurs on a time scale of a LOCA event for the NPM. Therefore, this requirement is satisfied by the design that precludes fuel temperature reaching critical heat flux and any significant fuel cladding heatup. For the NPM loss-of-coolant accident evaluation model core coverage and a minimum critical heat flux ratio are significantly greater than the safety limit, which precludes the occurrence of cladding oxidation (see Reference 7.2.1, Section 2.2). The NRELAP5 code and model have been assessed to experimental data to demonstrate the capability to reliably simulate the scenarios of interest. Conservative values for initial conditions and boundary conditions ensure an overall conservative analysis result.

2.1.2 Regulatory Guide 1.203

Regulatory Guide 1.203, “Transient and Accident Analysis Methods” (Ref: 7.1.3), describes a process that the NRC staff considers acceptable for industry use to develop and assess evaluation models used to analyze transient and accident behavior that is within the design basis of a nuclear power plant. An evaluation model is the calculational framework for evaluating the behavior of the reactor system during a postulated transient or design basis accident.

The containment response analysis methodology is an extension of the NuScale LOCA, valve opening event and non-LOCA methodologies developed following the guidance of Regulatory Guide 1.203. This report references the LOCA, valve opening event and non-LOCA methodologies and identifies and justifies the differences in the containment response methodology when compared to those methodologies.

3.0 Analysis

3.1 Modeling Software

The containment response analysis methodology uses the NRELAP5 system thermal-hydraulic code, which is a NuScale-modified version of the RELAP5-3D© v 4.1.3 code. NRELAP5 is used for all LOCA and non-LOCA transient and accident analyses, including the response of the CNV. The NRELAP5 simulation model used for the containment response analysis methodology is also similar to the NRELAP5 simulation models used for the LOCA, valve opening event and non-LOCA methodologies, which are presented in Reference~~s~~ 7.2.1 and Reference 7.2.2. The phenomena identification and ranking tables (PIRT) developed for the LOCA and non-LOCA methodologies are applicable to the containment response analysis methodology. The qualification of the LOCA and non-LOCA methodologies, in particular the comparisons to separate effects tests and integral effects tests, applicable to the containment response analysis methodology are presented in Section 4.1. The ~~differences in the~~ NRELAP5 simulation models used in the containment response analysis methodology as compared to the LOCA and non-LOCA models, along with the rationale for selection of conservative initial and boundary conditions, are the subject of this report.

3.2 NRELAP5 Base Simulation Model Development

3.2.1 RELAP5-3D©

RELAP5-3D©, version 4.1.3 was used as the baseline development platform for the NRELAP5 code. RELAP5-3D© was procured by NuScale and subsequently features were added to address unique aspects of the NuScale design and licensing methodology. The following is a brief description of the RELAP5-3D© code.

The RELAP5-3D© code has been developed for best-estimate transient simulation of light water RCSs during postulated accidents. The code models the coupled behavior of the RCS and the core for LOCAs and operational transients such as anticipated transient without scram, loss of offsite power, loss of feedwater, and loss of flow. A generic modeling approach is used that permits simulating a variety of thermal hydraulic systems. Control system and secondary system components are included to permit modeling of plant controls, turbines, condensers, and secondary feedwater systems.

The RELAP5-3D© code is based on a non-homogeneous and non-equilibrium model for the two-phase system that is solved by a fast, partially implicit numerical scheme to permit economical calculation of system transients. The code includes many generic component models from which general systems can be simulated. The component models include pumps, valves, pipes, heat releasing or absorbing structures, reactor kinetics, electric heaters, and control system components. In addition, special process models are included for effects such as form loss, flow at an abrupt area change, branching, choked flow, boron tracking, and noncondensable gas transport.

3.2.2 RELAP5-3D© Quality Assurance

NuScale Power procured RELAP5-3D© v.4.1.3 from the Idaho National Laboratory through a commercial-grade dedication process that complies with NQA-1-2008 and NQA-1a-2009 requirements. The commercial-grade dedication evaluation determined that verification of certain of the critical characteristics required testing. Eleven test cases were identified for verification, figures of merit, and acceptance criteria. Included were models of the NPM along with NuScale proprietary test programs, legacy tests, and special feature tests. These cases constitute the matrix for commercial-grade dedication acceptance testing as discussed by the LOCA evaluation model report (Reference 7.2.1), Section 6.1.2.

RELAP5-3D© v.4.1.3 was then placed under the NuScale quality assurance program as NRELAP5 Version 0.0. Subsequent NRELAP5 versions were developed and placed under the NuScale Quality Assurance Program including the technical code revisions listed in Table 3-3 along with code corrections and administrative code revisions.

3.2.2.1 NRELAP5

NRELAP5 is NuScale's proprietary system thermal-hydraulic computer code for use in engineering design and analysis. NRELAP5 was developed at NuScale, using RELAP5-3D© v.4.1.3 as the initial baseline. Chapter 6 of the LOCA Evaluation Model (Ref: 7.2.1) is a summary of the RELAP5-3D© code and the revisions incorporated by NuScale to produce the NRELAP5 code used in ~~both~~ the LOCA Evaluation Model, the Evaluation Model of the valve opening events (Reference 7.2.1, Appendix B), and the Non-LOCA Evaluation Model (Ref: 7.2.2). The new models in NRELAP5 are listed in Table 3-3 along with the application in the containment response analysis methodology.

Table 3-1 New NRELAP5 models

New Model	**Application in Containment Response Analysis Methodology**
Condensation heat transfer • {{ }}2(a),(c)	Used for condensation heat transfer on the CNV inside diameter and inside the decay heat removal system (DHRS) heat exchanger tubes
Critical flow • Moody critical flow model for two-phase flow conditions	Used for two-phase saturated critical flow
Helical coil SG component • Heat transfer correlation • Friction correlation	Used for modeling the helical coil SGs

3.2.3.2 NRELAP5 Non-Loss-of-Coolant Accident Evaluation Models

The NRELAP5 non-LOCA models are summarized in this section. The objectives of the NRELAP5 non-LOCA models are to analyze the spectrum of non-LOCA transients and accidents for the NuScale SMR, and to demonstrate compliance with the regulatory acceptance criteria.

3.2.3.2.1 **Inadvertent Operation of Emergency Core Cooling System**

The inadvertent operation of ECCS events include the inadvertent opening of an RVV or an RRV. Both events involve an initial primary system M&E release through the inadvertently opened valve into the CNV, and a subsequent actuation of the remaining ECCS valves that results in a second M&E release into the CNV. ~~The FSAR Section 15.6.6~~Reference 7.2.1, Appendix B describes the methodology for analyzing these events and is the starting point for developing the valve opening event models in the primary system containment response analysis methodology.

3.2.3.2.2 Secondary System Pipe Breaks

The NRELAP5 non-LOCA model is the starting point for developing the MSLB and FWLB models in the containment response analysis methodology. Figure 3-2 shows the non-LOCA NRELAP5 nodalization diagram.

Conservative modeling of the secondary pipe breaks to ensure a bounding M&E release includes the following elements:

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}}2(a),(c)

Containment Vessel and Reactor Pool Models

The CNV and reactor pool models for the MSLB and FWLB containment response analysis methodology are the same as the modeling for LOCA. Refer to Section 3.2.4.1.

3.3 Containment Response Analysis Methodology for Primary System Release Events

Section 3.3 presents the details of the containment response analysis methodology for primary system releases resulting from primary system breaks and valve opening events. The NRELAP5 computer code described in Section 3.2.2.1 and the LOCA containment response analysis model described in Section 3.2.4.1 are applied using the methodology in this section to meet the NRC regulations and regulatory guidance in Section 2.0.

3.3.1 Primary System Mass and Energy Release Methodology

3.3.1.1 Loss-of-Coolant Accident Scenario Phenomena Identification and Ranking Table Results

NuScale has performed and documented a PIRT for the LOCA scenarios resulting from primary system breaks and ECCS valve opening events. Loss-of-Coolant Accident Evaluation Model Report (Reference 7.2.1), Chapter 4.0, summarizes the LOCA phenomena identification and ranking table. The results of the LOCA phenomena identification and ranking table were used in the development of the NRELAP5 code, the NRELAP5 LOCA model, and the LOCA evaluation model. As discussed in Reference 7.2.1, Appendix B.7, there are no significant differences in physics phenomena between the LOCA and valve opening events for the NuScale NPM. Therefore, the high-ranked phenomena from the LOCA PIRT also apply to the valve opening events.

The results of the LOCA scenario PIRT are directly applicable to the primary system M&E release and resultant CNV pressure and temperature response that are the focus of the

4.0 Qualification and Assessment

4.1 Assessment of Methodology and Data

4.1.1 Primary System Release Effects Code and Model Qualification

The NRELAP5 code has been qualified or assessed to the separate effects and integral effects tests as described by LOCA Evaluation Model Report (Reference 7.2.1), Chapter 7.0 to demonstrate the capability to simulate LOCAs in the NPM. Reference 7.2.1 Appendix B describes extension of the LOCA EM for application to valve opening events. The results of the NRELAP5 comparisons to data establish the capability of the code to model the NPM design for the LOCA analysis. The most important assessment activities were those comparing to integral LOCA tests conducted in the NIST-1 facility.

The following two key known scaling distortions are relevant to the scope of the containment response analysis methodology:

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Neither of the above phenomena have an impact on the peak CNV pressure. The first distortion is addressed by the containment response analysis methodology by closure of the MSIVs. The second distortion is addressed by the overall conservative modeling of CNV heat transfer in the containment response analysis methodology, which includes use of conservative initial conditions and boundary conditions that are discussed in Section 3.4.

The LOCA Evaluation Model Report (Reference 7.2.1, Section 8.2) also presents the evaluation of the adequacy of the NRELAP5 code and LOCA Evaluation Model for modeling LOCAs in the NPM. ~~The following action was identified as needed to address adequacy issues relative to the containment response analysis methodology:~~

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No additional qualification activities were performed for the LOCA or valve opening containment response analysis methodology as the LOCA evaluation model qualification activities addressed in LOCA Evaluation Model Report (Ref: 7.2.1) are adequate.

4.2 Testing Results

4.2.1 NuScale Integral System Test Facility Testing

A scaled facility of the NPM was constructed at Oregon State University, referred to as the NuScale Integral System Test Facility-1, or NIST-1, facility, to assist in validation of the NRELAP5 system thermal-hydraulic code. The facility is designed to perform various tests, including LOCA tests. A detailed description of NIST-1, the NRELAP5 model of the facility, and the NRELAP5 validation testing, for the LOCA EM, is provided in Reference 7.2.1, Section 7.5.

The NRELAP5 predictions of CNV pressure, level and temperature documented in Reference 7.2.1 show good fidelity to NIST-1 experimental measurements as follows.

The CNV level and pressure response is predicted with reasonable to excellent agreement to RCS discharge line break experimental measurements as discussed by Reference 7.2.1, Section 7.5.6.

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The CNV pressure response is predicted with reasonable to excellent agreement to spurious RVV opening experimental data as discussed by Reference 7.2.1, Section 7.5.8.

A separate high pressure condensation test described by Reference 7.2.1, Section 7.5.4 demonstrates that NRELAP5 has the capability to predict condensation rates for various pressures with reasonable to excellent agreement to experimental data.

Reference 7.2.1, Appendix B.7 provides additional NRELAP5 assessment results for an updated spurious RVV opening test. The updated RVV spurious opening test provides better understanding of the impact of different ECCS orifice sizes on RVV opening events. The CNV pressure response is predicted with reasonable to excellent agreement.

Reference 7.2.1, Appendix C provides additional NRELAP5 assessment results for a spurious RRV opening event test. The CNV pressure response is predicted with reasonable agreement.

Enclosure 3:

Affidavit of Zackary W. Rad, AF-0119-64082

NuScale Power, LLC

AFFIDAVIT of Zackary W. Rad

I, Zackary W. Rad, state as follows:

1. I am the Director, Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.
2. I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:
    a. The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.
    b. The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.
    c. Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.
    d. The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.
    e. The information requested to be withheld consists of patentable ideas.
3. Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying Request for Additional Information response reveals distinguishing aspects about the method by which NuScale develops its containment response analysis.

    NuScale has performed significant research and evaluation to develop a basis for this method and has invested significant resources, including the expenditure of a considerable sum of money.

    The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

    If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

4. The information sought to be withheld is in the enclosed response to NRC Request for Additional Information No. 465, eRAI No. 9494. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.
5. The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).
6. Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:
   a. The information sought to be withheld is owned and has been held in confidence by NuScale.
   b. The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.
   c. The information is being transmitted to and received by the NRC in confidence.
   d. No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.
   e. Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 9, 2019.

Zackary W. Rad