U.S. NUCLEAR REGULATORY COMMISSION

AUDIT SUMMARY REPORT FOR THE REGULATORY AUDIT OF NUSCALE POWER'S CONTAINMENT AND VENTILATION SYSTEMS AS PART OF THE DESIGN CERTIFICATION APPLICATION REVIEW

APRIL 3, 2017 THROUGH JULY 2, 2019

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I. AUDIT LOCATION AND DATES

The U.S. Nuclear Regulatory Commission (NRC) staff conducted the audit from NRC Headquarters in Rockville, Maryland, through NuScale Power, LLC (NuScale's) electronic reading room (eRR) and at (1) the NuScale Office at 11333 Woodglen Drive, Suite 205, Rockville, Maryland 20852 and (2) NuScale Integral System Test (NIST-1) facility located at Oregon State University in Corvallis, Oregon. This summary report is for the audit conducted from April 3, 2017, through July 2, 2019.

II. BACKGROUND AND AUDIT BASIS

In a letter dated December 31, 2016, NuScale submitted to the NRC Revision 0 of the NuScale Standard Plant Design Certification Application (DCA) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17013A229). The NRC staff initiated this DCA review on March 27, 2017. NuScale submitted Revisions 1 and 2 of DCA to the NRC on

March 15 and October 30, 2018 (ADAMS Accession Nos. ML18086A090 and ML18311A006). Application documents for the NuScale design are available at the NRC Website at https://www.nrc.gov/reactors/new-reactors/design-cert/nuscale/documents.html.

The intent of this audit, in part, was to gain a more detailed understanding of the NuScale design in technical areas associated with containment and ventilation systems and identify information that will require docketing to support the basis of the regulatory decision. The NRC staff issued an audit plan on March 29, 2017, and addenda on June 28, 2017 (ADAMS Accession Nos. ML17087A077 and ML18177A087, respectively). The addenda narrowed the audit scope by limiting it to three specific areas: containment pressure analysis, containment integrated leakage rate testing, and containment Isolation.

The staff issued an interim audit summary report on November 9, 2018 (ADAMS Accession No. ML18291B228). This audit summary report has been prepared in accordance with NRO-REG-108, “Regulatory Audits,” Revision 0, April 2, 2009 (ADAMS Accession No. ML081910260).

III. DOCUMENTS AUDITED

The NRC staff performed an audit of documents, as necessary, shown in the Enclosure.

IV. AUDIT ACTIVITIES AND SUMMARY OF FINDINGS

The audit activities included the following review areas:

- containment peak pressure analysis;
- containment heat removal;
- methodology of mass and energy release from the reactor coolant system;
- ASME qualification of the containment vessel;
- containment leak rate testing, including NuScale exemption request no. 7, “10 CFR 52, App. A, General Design Criteria (GDC) 52 Containment Leakage Rate Testing”;
- combustible gas control, including NuScale exemption request no. 2, “10 CFR 50.44(c)(2) Combustible Gas Control”;
- equipment survivability;
- containment isolation, including NuScale exemption request no. 9, “10 CFR 50, Appendix A, GDC 55, 56, and 57 Containment Isolation”;
- containment flooding and drain;
- pipe break hazard analysis;
- NIST-1 test observation;
- range of reactor recirculation valve opening;

- bioshield redesign
- containment stratification; and
- containment nodalization.

The NRC staff had numerous (more than 100) interactions with NuScale, including teleconferences, test observations, and face-to-face meetings. The audit enhanced the NRC staff's review of the NuScale DCA by providing an opportunity to review non-docketed supporting design information that was not submitted with the DCA. Specifically, the audit documents that were made available to the NRC staff for review have enabled the staff to efficiently obtain the information that they need to continue their review and prepare their safety evaluation report. Furthermore, the audit allowed the NRC staff to limit issuing requests for additional information (RAIs), as needed, to:

- Gain a better understanding of the detailed calculations, analyses and/or bases underlying the formal application and confirm the staff's understanding of the NuScale application.
- Identify additional information, necessary for the applicant to supplement its application, assisting the staff to reach a regulatory decision.
- Establish an understanding in an area where the staff has identified potential concerns, and in turn allow the staff to issue clear RAIs enabling the applicant to provide quality and timely responses.
- Enhance the staff's understanding of the NuScale design in support of making a regulatory decision.

Specific audit activities and summary findings are provided below.

VI.A Containment Analysis

The present document provides a summary of the containment and ventilation audit that was performed in support of the staff review of the NuScale Final Safety Analysis Report (FSAR) Section 6.2.1.1 on Containment Structure and the Containment Response Analysis Methodology (CRAM) technical report (TeR) (Reference 1) that is incorporated by reference. The review was mainly related to the containment thermal-hydraulic design and containment pressure and temperature response that would result from the limiting containment design basis event. Due to the novel nature of the NuScale design and the phenomenological complexities involved, the staff review of the containment thermal hydraulic design required a team effort of several reviewers from the Office of New Reactors, Office of Research, and contractors from ERI and NUMARK. The review also required a close collaboration among the Chapter 6 and loss-of-coolant accident (LOCA) topical report (TR) (Reference 2) reviewers due to several overlapping review areas that involved several audit activities.

Audit Scope

NuScale uses the NRELAP5 code, a custom version of the RELAP5-3D baseline code, to perform the integrated nuclear power module (NPM) design basis safety analysis of the reactor pressure vessel (RPV) mass and energy (M&E) release during the blowdown and the resulting containment pressurization and heat removal to the cooling pool with enough margin to meet

the regulatory requirements. The CRAM is an extension of the NuScale LOCA, valve opening event, and non-LOCA methodologies. The CRAM TeR (Reference 1) references both LOCA and non-LOCA methodologies and justifies any differences from those methodologies, such as the initial and boundary conditions changes needed to conservatively bias the M&E release to maximize the containment pressure and temperature response. Table 3-3 in the CRAM TeR describes five cases of primary system M&E release used to calculate the peak containment vessel (CNV) pressure and wall temperature.

The staff audited the applicant's sensitivity cases of the peak containment pressure and temperature, to investigate several key factors such as the condensation heat transfer modeling on the containment inside surface; effect of noncondensable gases; natural convection heat transfer modeling; nodalization of containment volume and heat structures; and liquid thermal stratification effects inside the containment as well as in the cooling pool. Below is a summary of the related audit activities. In general, the peak containment pressure and temperature were found to not be very sensitive to these factors to significantly reduce the safety design margins.

Conservatisms in the NPM CNV Model/Initial & Boundary Conditions

As discussed below, the staff audited conservatisms used in the applicant's NRELAP5 modeling decks for initial and boundary conditions for the spectrum of M&E release events to ensure that the NPM containment design basis safety analyses were conservative.

Effect of Reduced Pool Level on Peak CNV Pressure and Wall Temperature

The ultimate heat sink (UHS) pool structure has several connections at 60 ft elevation that are not seismic Category I design. In addition, the dry dock gate design is not seismic Category I, either. Failure of the dry dock gate (if the dock is empty) would cause the nominal 68 ft pool level to drop by 12 ft to the 56 ft level. Therefore, it is necessary to justify how the UHS pool would maintain the required water level if these non-seismic Category I components were to fail. To justify that assuming a higher (i.e., 65 ft) reactor pool water level is only required as an initial condition prior to the initiation of the event, a sensitivity study was used to show that with UHS pool water level at 55 ft (immediately after the initiation of the event) the CNV peak pressure would remain within its safety limits.

As described in ECN-A013-6778 R0.pdf, "Additions to Support Resolution of 10/15/2018 NRC Comments" (Document 111), NuScale ran a sensitivity case in which the initial pool level was reduced to 55 ft while the initial CNV wall temperatures above the 55 ft elevation remained the same as that when crediting the 65 ft pool level during the steady state run. Specifically, the limiting peak CNV pressure case was re-run with the pool level initialized at 55 ft instead of 65 ft and with CNV wall temperatures above 55 ft initialized at the same values as the original limiting case (i.e., crediting the 65 ft pool level for CNV wall temperature initialization only). The sensitivity analysis showed that by using a 55 ft initial pool level, the maximum CNV internal pressure rose from 986.1 psia to 986.4 psia, and the maximum wall temperature rose from 492.3 to 493.3 °F, which the staff finds as not safety significant.

Effect of Reduced CNV Free Volume on Peak CNV Pressure and Wall Temperature

The staff noted that Revision 1 of EC-A013-2341 (Document 26), which describes the nominal NRELA5 base case to which each of these individual analyses are performed, used [redacted] margin in CNV free volume that was reduced to [redacted] in Revision 2 (Document 25). An audit of the applicant's containment design basis event (DBE) response analysis document

(EC-A013 2341, Revision 2) shows that the free volume in NPM containment is approximately [redacted] which is about [redacted] of the nominal containment free volume of [redacted] The [redacted] reduction in containment free volume was conservatively assumed to account for blockage in containment by components, such as, piping, etc. This was in line with Tables 3-5 and 5-1 of the CRAM TeR (Reference 1) indicating that the nominal CNV free volume is reduced by [redacted] as a conservative initial condition for the containment response analysis to account for uncertainty in design. The staff issued RAI 9482, Question 06.02.01.01.A-18 asking the applicant to justify the [redacted] volume adjustment used with an outdated NRELAP5 base model.

NuScale re-evaluated the containment free volume by incorporating containment geometry changes and an allowance for RCS thermal expansion, piping, valves, cabling and miscellaneous components such as platforms and ladders. With these changes the net containment free volume increased to [redacted] Accordingly, the applicant used [redacted] as the bounding minimum CNV free volume value in the containment response analysis. To show that the CNV peak pressure will remain within its safety limits even when assuming the more conservative CNV free volume, NuScale performed a sensitivity study in which the CNV free volume of was reduced by [redacted] The results show an increase of approximately [redacted] from the limiting case when assuming CNV free volume at [redacted] This demonstrates that with a further [redacted] CNV free volume reduction to account for uncertainties in estimation and manufacturing, the CNV peak pressure would lose [redacted] peak pressure margin that would not be significant. NuScale's reevaluation of the containment free volume has addressed the staff concerns that the base NRELAP5 model described in [redacted] with ECN.pdf, "Reactor Module NRELAP5 Model - with ECN" (Document 113) had not been previously updated for geometry changes reported in [redacted]_System_Transient_Model_Input_Parameter_Calculations.pdf, "System Transient Model Input Parameters Calculation" (Document 114). NuScale stated that it planned to use a minimum containment free volume value of [redacted] to account for RCS thermal expansion and an allowance for piping, valves, cabling and miscellaneous components such as platforms and ladders (RAI 9482, Question 06.02.01.01.A-19).

[redacted] is the updated CNV response analysis that is the basis for the CNV RAI responses such as RAI 9482, and [redacted] documents the sensitivities discussed in the response to RAI 9482, Question 06.02.01.01.A-18 (e.g., pool level sensitivity). Appendix B at the end of [redacted] (Document 112) includes supplemental sensitivities and was modified but it did not change any of the results already documented in [redacted] This appendix includes the description and results of three sensitivities added to Appendix B to support the resolution of NRC questions asked during the October 15, 2018, teleconference.

<u>High Initial CNV Internal Pressure of 3.0 psia</u>

The applicant had indicated that the initial CNV pressure was unintentionally reset to 1.0 psia at the start of the transient run instead of 2.0 psia as listed in the CRAM TeR Tables 3-5 and 5-1. The updated containment response analysis of record resolves this error by correcting the CNV initial conditions to maintain a consistently high CNV internal pressure of 3.0 psia, as discussed in the RAI 9482, Question 06.02.01.01.A-18 response, for the steady state initialization leading to the event initiation. The figures included in the RAI 9357, Question 06.02.01.01.A-4 response show that steady state initial conditions, including a steady CNV internal pressure of 3.0 psia, are successfully reached during the CNV analysis model steady state runs prior to the initiation of primary and secondary transient events at time zero reducing the likelihood of unintentionally

re-initializing a model parameter. The response also indicates that the CRAM TeR Figure 5-31 has been revised as shown in the markup provided in the RAI response. The staff audited EC-0000-5435-R0_final.pdf, “Containment Pressure Initial Condition Sensitivity Calculations” (Document 20) that documents the sensitivity cases of 2 psia, 3 psia, and 4 psia of initial containment pressure for peak CNV pressure in support of RAI 8793, Question 06.02.01-2. The staff agrees that the applicant was able to show that 2 psia initial containment pressure is adequate. Therefore, using 3 psia is conservative that builds about 2 psia conservatism in the resulting peak containment pressure. The peak wall temperature is not very sensitive to the initial containment pressure.

Effect of Noncondensable Gases on Condensation Heat Transfer (Timing & Composition)

The staff reviewed and audited the effect of noncondensable gases (NCG) on condensation heat transfer and the peak CNV pressure and wall temperature. The noncondensable gases typically included hydrogen, oxygen, and nitrogen. NuScale used a noncondensable gases mass of 65.7 lb. initially present in the containment at 3 psia initial pressure during normal operation. Later, while addressing the staff RAIs, the applicant made a modeling change to incorporate the effect of an additional 65.4 lb. mass of noncondensable gas release from RPV into CNV during the transient. NuScale’s response to RAI 9482, Question 06.02.01.01.A-18, dated October 26, 2018, described the modeling change and the CNV response analysis that included the impact of additional noncondensable gases that existed in the RPV as dissolved in reactor coolant and as vapor in the pressurizer (PZR) gas space. The noncondensable gases that are discharged from RPV to CNV include residual hydrogen used for the RCS water chemistry, NCG accumulated in the upper pressurizer during extended period of reactor operation, and the NCG stripped from the reactor coolant during the steam flashing. The staff needed to know the NCG composition and whether the additional 65.4 lb. of NCG was conservative. The additional noncondensable gas release was not accounted for in the spreadsheets submitted with the October 26, 2018, RAI 9357 response. Due to insufficient analysis details provide in the DCA, the noncondensable gases composition and release rate were discussed during audit teleconferences held on February 14 and 22, 2019, in the NuScale’s Rockville office. The staff asked for the assumed composition (e.g., hydrogen, nitrogen, air), release location and release start time, and M&E release rates of the additional noncondensables into the CNV during the transient.

NuScale performed a sensitivity study investigating the impact of the 65.4 lb. NCG release from the RPV to CNV in addition to the 65.7 lb. pre-existing in the CNV during normal operation, on CNV peak pressure and temperature results that was documented in ECN-A013-7140-R0.pdf, “Additions to Support Resolution of 1/16/19 NRC Comments” (Document 115). ECN-A013-7140-R0 shows that estimating [redacted] hydrogen as a conservative mass fraction of the total RPV noncondensable mass of 65.4 lb. The results show an increase in peak CNV pressure and wall temperature of [redacted] when assuming an RPV noncondensable mass composition of [redacted] air and [redacted] hydrogen, with respect to the limiting design-basis case of 100 percent air for the additional noncondensables for the analysis of record. However, the document did not provide any details of 65.4 lb. as a conservative estimate of the total RPV noncondensable mass release to the CNV. Following the April 16, 2019, audit call, NuScale posted EC-A010-5539_Rev0.pdf, “RXM Noncondensable Gas Mass” (Document 116) in the eRR to describe the methodology to calculate the mass and composition of noncondensable gases present in the CNV during normal operation (65.7 lb.) as well as in the RPV and released into CNV during the transient at the ECCS actuation (65.4 lb.). The analysis assumed that the total mass of additional noncondensables existing within the RPV during operation and releasing into the CNV is composed of hydrogen, oxygen, and

nitrogen. The audit showed that the mass of hydrogen in the RPV is calculated as the sum of the dissolved hydrogen in the RCS liquid and the hydrogen gas accumulated in the PZR. The volumetric and partial pressures information was converted to mole fractions using the Henry's law. The staff also audited EC_B020_4365_R0, Containment Gas Composition Calculation.pdf, "Containment Gas Composition Calculation" (Document 10), which provides additional details on the calculation of the composition NCG that releases into the CNV during design basis and beyond design basis events. The staff also asked for RCS chemistry report and NuScale uploaded ER-A030-3027 R0 RCS Chemistry Report.pdf, "ER-A030-3027 – RCS Chemistry Report" (Document 117) for audit. This document describes the reactor coolant chemistry parameters during start-up, normal operation, and shutdown. Gaseous hydrogen at high temperature is added to the reactor coolant through the CVCS circulation loop to maintain a [redacted] H_2O dissolved hydrogen content to ensure a reducing environment to scavenge the oxygen produced by radiolysis.

EC-A010-5539_Rev0 (Document 116) documents two cases to calculate the NCG mass using different assumptions for PZR pressure [redacted]), PZR level, RCS cold temperature, and PZR NCG partial pressure, as tabulated in Table 3-1 in the document. Even though Case 1 leads to a much higher total NCG in the NPM [redacted]), its PZR level is unrealistically low at [redacted] so the staff concluded that Case 1 is overly conservative. The staff found that Case 2 with 65.4 lb. NCG generation in the RPV (or 131.1 lb. total NCG in the RPV and CNV) at [redacted] PZR level as the bounding case with enough conservatism. The applicant justified [redacted] PZR NGC partial pressure [redacted]. The applicant used [redacted] for calculating the noncondensables in Case 2, rather than [redacted] Varying the PZR pressure from [redacted] would increase the noncondensable gases (NCGs) partial pressure above [redacted] As the application of heating increases the pressure of the PZR that is a saturated steam environment, the PZR liquid temperature would also increase. As the increased PZR liquid temperature would correspond to a higher corresponding vapor pressure, according to Henry's law, it will also lead to a higher Henry's Law constant that will lead to a higher partial pressure of the noncondensables in the PZR vapor space for the same concentration of the dissolved NCGs in the RPV water. This would essentially mean a higher mass and concentration of NCGs in the PZR vapor space. However, the staff does not expect a significant change in the noncondensables mass. So, the use of a lower [redacted] is not a cause of concern about calculating the noncondensable's mass. Besides, the applicant has used the higher [redacted] as the input condition for the DBA CNV M&E release transient analysis. The staff also found that the NCG release initiation would coincide with the ECCS actuation and not the RRV opening, as there is no credible physical mechanism to support the release of NCG mainly that is mainly accumulated in the PZR, from RPV into the CNV without RVV opening. That is why the release of the NCGs at the elevation of the RVVs on top of the RPV at PZR initial pressure and temperature conditions is acceptable. The staff concludes that conservative assumptions, such as using a lower pressurizer level than the limiting transient and higher initial containment pressure, were used to maximize the noncondensable mass in the CNV during operation and released into the CNV from RPV during the DBE. Further, NuScale's sensitivity studies showed a small impact of noncondensables composition and release rate on peak CNV pressure.

The staff audited the above information supporting NuScale's design change involving the release of additional noncondensables on the CNV peak pressure and performed confirmatory calculations to understand the sensitivity of the CNV peak pressure and temperature to the released noncondensables. The staff MELCOR confirmatory calculations confirmed about [redacted] impact of [redacted] Hydrogen release from the RVV location regardless of the discharge time, which is consistent with [redacted] impact predicted by the applicant. The staff

also found it acceptable that the RPV conditions used for estimating the RPV noncondensable mass were not used for the NRELAP5 CNV pressure response safety analyses, as they would have been less conservative with respect to peak CNV pressure.

Condensation Heat Transfer Modeling on the CNV Inner Surface including the Effects of Noncondensable Gases

The CRAM models steam condensation on the containment inner surface using the [redacted] (Reference 4) that was added in NRELAP5 as described in LOCA topical report (Reference 2). The original Shah condensation correlation that was published in 1979 (Reference 5) and included in RELAP5-3D, has the same form as that of Dittus Boelter correlation (Reference 6), except that the Reynolds number is based on the liquid part of the two-phase flow, which requires a vapor quality multiplier to the total-flow based Reynolds number. Finally, the condensation heat transfer Nusselt number becomes a function of liquid Nusselt number, mass quality, and the reduced pressure. The [redacted] which involves the liquid and vapor viscosities and the reduced pressure. The [redacted] uses the vapor volumetric flux to determine laminar, transition, or turbulent flow of the film, in contrast to a Reynolds-number-based criterion used for the Nusselt's classical gravity-assisted condensate film flow modeling. The [redacted] was added to NRELAP5 and is described in LOCA TR (Reference 2).

The staff was concerned about the applicability of the [redacted] to the NuScale containment design and geometry and requested the applicant to justify its use. One concern was that the [redacted] was developed based on in-tube condensation data taken on small diameter tubes (2 - 49 mm), so its application to the NuScale containment's substantially larger diameter with a large annular flow area was outside the range of applicability. The implementation and working of the NRELAP5 wall condensation model in the LOCA TR and the NRELAP5 theory manual was also not clear to the staff (Document 118).

The staff was not able to assess the conservatism in the NRELAP5 condensation heat transfer model and the sensitivity of the predicted pressure and temperature margin to condensation modeling. Therefore, the staff sought justification from the applicant for using the [redacted] for condensation heat transfer modeling, as discussed during audit calls held on January 19 and February 2, 2019, and the February 2, 2019, audit meeting held at the NuScale's Rockville office. In response, NuScale conducted a study that evaluated the impact of using the [redacted]

NuScale made the details of the study and results available for audit by adding NCI-0119-64169-R0_DRAFT_NRC.pdf, "NRELAP5 Implementation of a Vertical Plate Condensation model" (Document 119) to the eRR. The staff audited the document that showed that the [redacted] is conservative with respect to CNV pressure for both NPM as well as NIST-1 HP-49 test CNV geometry. The staff determined that NuScale had made available for audit a high-level summary of the implementation and benchmarked the results of

the [redacted] used in the peak containment pressure design basis analyses.

The FSAR, CRAM TeR (Reference 1), and LOCA TR (Reference 2) mention that the effects of NCGs were accounted for on condensation on CNV inside diameter. NCI-0119-64169-R0_DRAFT_NRC (Document 119) also describes the Colburn-Hougen iteration method (Reference 8) that is used in NRELAP5 to account for the adverse effect of noncondensable gases on the condensation heat transfer coefficient, regardless of the condensation correlation employed. The Colburn-Hougen method is used to model the deterioration of the interfacial heat transfer on the liquid film condensing on the containment wall, due to the insulating effect of NCG present in the mixture, on the heat transfer between the vapor/gas and the wall. Without noncondensable, the interface temperature is the volume saturation temperature. In the presence of noncondensable gases, the Colburn and Hougen diffusion method is used to iteratively solve for the interface temperature between the liquid film and the vapor volume. Here the saturation temperature is substituted for by the iterated interface temperature as described in the NRELAP5 Theory Manual (Document 118) Section 2.6.2.6.5.

The Colburn-Hougen iteration is based on the principle that the total heat transferred by condensing vapor to the liquid-vapor/gas interface by diffusing through the noncondensable gas film plus the heat transferred by sensible heat on the film surface is equal to the heat transferred through the condensate. This has the effect of modifying the average film temperature and properties at this temperature. The condensation heat flux to the wall depends on the degree of wall subcooling relative to the saturation temperature based on the partial pressure of the vapor and other factors such as the liquid film thickness, turbulence, vapor/gas shear, etc. The heat released at the vapor/gas-liquid interface is transferred through the liquid film and into the wall. The applicant did not change NRELAP5 for the standard Colburn-Hougen method that was already coded in RELAP5. Following a LOCA, the containment pressure increases rapidly, reducing the noncondensable gas concentration in the containment making the effects on noncondensables less important.

The audited material (Document 119) showed that the [redacted] calculated peak containment pressure results are slightly more conservative than that of the [redacted]. The peak containment pressure calculated for the analysis of record using the [redacted] which is slightly higher than [redacted] that is calculated by using the [redacted] when applied to the entire containment vertical surface. The [redacted] peak pressure prediction took no credit for the [redacted], which is conservative. The staff also noted that the [redacted] Document 119 also demonstrated similar CNV pressure conservatism predicted for the NIST-1 HP-49 test CNV geometry by using the [redacted] vis-à-vis [redacted]. So, the staff did not find much sensitivity of peak pressure to switching from one condensation correlation to the other, as well as switching from full-scale NPM design to NIST-1 scaled facility. This result is consistent with the staff's MELCOR and TRACE confirmatory calculations that use completely different condensation correlations but predict peak containment pressure and temperature that are comparable, though less conservative, to the ones predicted by NRELAP5. The staff found the audited information supporting the safety finding with respect to overall condensation heat transfer modeling.

During the February 22, 2019, audit meeting at NuScale's Rockville office, the staff asked for a high-level summary of recent implementation of the [redacted] on the containment wall showing that the [redacted] is more conservative with respect to CNV peak pressure. In response, NuScale supplemented its response to RAI 9494, Question 06.02.01.01-A17 on April 8, 2019, that providing a summary of the sensitivity study comparing the CNV peak pressure results with the [redacted] in NRELAP5. Based on its audit of the condensation heat transfer correlation and the treatment of the noncondensables effect on condensation heat transfer as documented in the above, the staff finds that the NuScale licensing basis containment analyses are based on a conservative condensation heat transfer modeling for the containment wall.

The staff completed the audit for NCGs and finished the review of the related information in the DCD, CRAM TeR, LOCA TR, and RELAP5 Theory Manual v1.3 about (1) mass of NCGs present initially in the CNV (2) mass of NCGs released from RPV to CNV, and (3) the Colburn-Hougen iteration method to account for condensation with NCG. The staff has no outstanding concerns about the NuScale's methodology to account for the impact of noncondensables on the condensation heat transfer.

RRV/RRV Opening Time Sensitivity

It was expected that a faster ECCS valve opening would result in a higher mass flow through the opened valve resulting in a faster RPV depressurization and possibly a different peak containment pressure. In response to RAI 9536 that was issued during the LOCA TR review, NuScale presented a sensitivity study using [redacted] ECCS valve opening times. The results summarized in the RAI response Table B-18 confirm that the faster ECCS valve opening is more conservative for minimum critical heat flux but shed no light on its impact on the calculated peak containment pressure.

The CRAM TeR (Reference1) does not discuss the valve stroke times. However, NuScale FSAR Part 2, Tier 2, Section 6.3 states that the "RRVs fully open in 10 seconds." However, an audit of EC-A010-1782, Revision 1, NRELAP5 Base model.pdf, "NRELAP5 Base model" that describes the NRELAP5 base deck (Document 120) showed a fast valve opening time of [redacted] used in the licensing basis NRELAP5 model that is consistent with the value used in the licensing basis NRELAP deck. The document also states that "It is assumed that fast valve opening time is limiting," which may be valid for minimum critical heat flux but might not be valid for peak containment pressure. Therefore, the staff asked and NuScale perform a sensitivity study of the peak containment analysis to ascertain whether a conservative RRV opening time had been used for the limiting transient. In response, NuScale reported insignificant sensitivity of the RRV opening time to the peak containment pressure. The staff audited this calculation and agreed that the containment peak pressure was not sensitive to the valve opening times.

Justification for the Initial Containment Wall Temperature Distribution Used for the Peak CNV Pressure Safety Analysis

The staff discussed with NuScale the containment heat-up under radiation heat transfer during normal operation as it needed to make a safety finding about the overall initial containment wall temperature distribution that was used calculate the CNV pressure transient response. In this regard, wall temperature initialization was discussed during audit calls held on January 19 and February 4, 2019, and the February 22, 2019, audit meeting in the NuScale's Rockville office. The staff asked for justification for the initial CNV wall and head temperature distribution used to simulate the limiting transient because it was not clear to the staff how NuScale credited the

110 °F *bulk* pool water temperature to initialize the CNV wall temperature for normal operation for the peak CNV pressure analysis, and whether the impact of heat loss under normal operation was considered across the annular CNV space to heat-up the CNV wall and pool water in the vicinity of the containment. The impact of natural convection-based pool water mixing (or lack thereof) and heat up near the containment also needed to be accounted for, as it could impact the CNV peak pressure even within 100 sec. The applicant needed to justify the CNV wall and head temperature distributions, which is the key to the condensation heat removal from the containment to mitigate the CNV pressure.

In response, NuScale added a document entitled "2-4-19 discussion input" (Document 121) to the eRR to provide justification for the initial containment wall temperature distribution. NuScale also added ECN-A013-7140-R0.pdf, "Additions to Support Resolution of 1/16/19 NRC Comments" (Document 115). In addition, NuScale added ECN-A010-7135_Rev0_2.pdf, "ECN – CNV Head Temperature Figures" (Document 122) to the eRR, which documents the CNV wall temperature distribution information referenced in the "2-4-19 discussion input" file (Document 121). As asked by the NRC staff in the January 16, 2019, public meeting, NuScale provided in Document 122 the colored depictions of the 2-D temperature distributions throughout the CNV head down to the pool water level, as extracted from the ANSYS finite element model results for the steady-state reactor operation before the transient. The staff had asked about the location of the minimum and maximum temperatures in the CNV head and found the 2-D colored plots to be consistent with the minimum and maximum CNV head temperatures tabulated in Table 5-3 of EC_A010_2686_00_RXM_Outline_Model_Calculation with ECNs.pdf, "Reactor Module Outline Model Calculation - with ECNs" (Document 123). ECN-A013-7140 (Document 115) and ECN-A010-7135 (Document 122) provide justification for the initial CNV wall temperature distribution and pool side heat transfer modeling. This issue was tracked in the SER under OI 6.2.1-2 and was partly related to RAI 9482 Question 06.02.01.01.A-18.

Based on its audit of ECN-A010-7135 (Document 122) and ECN-A013-7140 (Document 115) and review of NuScale's response to RAI 9482, Question 06.02.01.01.A-18, the staff accepted the procedure for the CNV wall temperature initialization and the resulting initial CNV wall temperature distribution.

The staff also audited ECN_A010_4781_00_Update Maximum Pool Temperature.pdf, "Add an Additional Pool Temperature Case" (Document 125) that summarizes a pool temperature sensitivity study for the pool temperature range of 40 °F – 120 °F. Results are summarized in Table 5-2 (Steady State Load Cases) and Table 5-3 (Heat Loss Rate and CNV Head Temperature for Steady State Condition) are updated. The document also includes an additional case run with a [redacted] pool temperature to establish a bounding maximum CNV temperature (Thermal only, bounding hot pool) that leads to the head temperature between [redacted] However, as the pool temperature technical specification is 110 °F, the [redacted] pool temperature run is not applicable to the design-basis accidents. Further, as audited in ECN-A013-6778 R0.pdf, "Additions to Support Resolution of 10/15/2018 NRC Comments" (Document 111), increasing the pool water temperature from 110 °F to [redacted] for the limiting case increased the maximum CNV pressure from [redacted] and maximum CNV wall temperature from [redacted]

Justification for the Natural Convection Heat Transfer Modeling

Per the CRAM TeR, natural convection heat transfer on the containment inside annular surface and outside vertical surface to the liquid pool are modeled using [redacted]

(Reference 9). No mixing is modeled in the reactor pool. The staff needed to ensure an appropriate use of the [redacted] which calculates an average heat transfer correlation coefficient applied from the top to bottom of the containment.

In natural convection, heat transfer is driven across the boundary layer that would start at the bottom of the containment as a laminar boundary layer and transition into a turbulent boundary layer at the upper part of the CNV surface that is exposed to high temperature steam condensation on the inside. As natural convection heat transfer coefficient increases with temperature difference between the surface and the bulk pool liquid, it would vary along the depth of the pool. It would be higher near the bottom of the CNV (where not much heat transfer takes place) and low near the top of the CNV wet area (where the condensation heat transfer contributes the most to the heat removal and CNV pressure mitigation). Therefore, using an average natural convection heat transfer coefficient could be non-conservative with respect to the CNV wall temperature initialization and peak pressure calculation. Therefore, during the audit, the staff requested justifications for the pool-side natural convection heat transfer modeling for the limiting transient. NuScale needed to demonstrate that the combination of highest condensation and lowest natural convection heat transfer coefficients near the top of the pool, and vice versa near the bottom of the containment is sufficiently representative with respect to the CNV wall temperature initialization and peak containment pressure calculation. NuScale provided justification for the pool side heat transfer modeling in "2-4-19 discussion input" (Document 121). It compared the natural convection heat transfer coefficient predicted by the [redacted] and the measured natural convection heat transfer coefficients for HP-02 Run1 test for the inside and outside heat transfer plate surfaces at the NIST-1 test facility. As shown on a previously undocketed HP-02 figure included in the document, the difference between the measured and predicted natural convection heat transfer coefficients for the HP-02 Run 1 containment on the same elevation along the pool depth inside and outside containment surfaces were insignificant to warrant a safety concern. NuScale also provided snapshots of the CNV outer diameter wall temperature in time as Figure 14, which shows that the effect of transient was not felt much on the CNV outer surface within the peak pressure time scale, and the natural convection heat transfer did not play a significant role in dictating the peak containment pressure. This conclusion was further corroborated by the RAI 9380, Question 06.02.01.01.A-5 response dated March 14, 2019, that showed that increasing the number of nodes along the reactor pool depth from 3 to 28 impacted the CNV peak pressure by [redacted] Therefore, no more justification is needed for using a constant pool-side natural convection heat transfer coefficient rather than the natural convection heat transfer coefficient distribution on the CNV outside surface along the pool depth. The staff found the information audited relative to the validation of the pool-side natural convection heat transfer coefficient to be responsive to the issue that does not merit further investigation.

The staff also audited an evaluation included as an appendix in ECN_B020_4991_R0 Add Evaluation of Hydrogen Mixing during ECCS operation.pdf, "Add Evaluation of Hydrogen Mixing during ECCS operation" (Document 37) to substantiate the assumption that during the emergency core cooling system operation, steam and hydrogen are well mixed and distributed throughout the containment vessel. The natural convection forces of steam and hydrogen in the annular containment vessel cavity of the reactor module (above the condensate liquid level) are evaluated within this appendix. This evaluation is performed to determine if steam and hydrogen movement due to natural convection during ECCS operation up to 72 hours is enough to characterize the regions as well mixed and distributed within the volume. Figure B-1 describes the evaluated containment region. A Rayleigh number analysis is used to establish that natural convection flow would be turbulent in the annular region between the heated RPV and the reactor pool cooled CNV. The staff agrees with the NuScale conclusion that the

introduction of some hydrogen into the steam environment does not significantly affect the natural convection flow regime.

Containment Nodalization Studies for the NPM & NIST-1 Test Facility to Demonstrate Sufficient Fidelity for Thermal Stratification Inside the Containment Liquid

The staff needed a greater understanding to assess the safety significance of the thermally stratified water in the CNV of the NPM during blowdown out to the time of peak containment pressure. Therefore, the staff requested NuScale to demonstrate enough fidelity in the safety analyses to predict thermal stratification and subcooling of the liquid water that would accumulate in the lower containment due to the blowdown and condensation of the flashing steam on the containment wall. The staff was concerned that overpredicting the containment liquid water temperatures due to nodal stratification could lead to an underprediction of the containment pressure. So, the staff needed a demonstration of lack of sensitivity of the simulation results to containment and liquid volume nodalization to establish NRELAP5's capability to accurately predict the containment pressure and temperature response and the peak containment pressure for the limiting DBE of the inadvertent opening of the RRV, which is the largest liquid space discharge event for the NuScale design.

NuScale Response to RAI 9380, Question 06.02.01.01.A-5, dated October 25, 2018, stated that "The updated containment vessel (CNV) response analysis evaluated effects of using a set of coarser and finer axial nodalization for the CNV volume, a finer reactor pool nodalization, and a finer CNV heat structure radial nodalization to determine the most limiting nodal representation with respect to CNV peak pressure and temperature." However, the NuScale RAI response did not provide any results from the nodalization study, but states that "the fine axial nodalization model resulted in an increase in peak pressure and peak temperature." No detailed assessment was included in the RAI response to Question 06.02.01.01.A-5. It was not clear to the staff how NuScale determined that [redacted] were enough. The staff raised this concern with NuScale during the audit calls, audit meeting, and public meetings on January 1, February 4, 14 and 22, 2019. NuScale needed to show that that the finer nodalization is appropriately refined, specifically in the liquid region, and a further finer nodalization would not make a significant impact on the peak CNV pressure margin. The staff suggested NuScale to include comparison plots for the three axial nodalization schemes [redacted] in the supplemental RAI response, on the same respective graphs.

NuScale agreed to perform containment nodalization studies for the NPM as well as the NIST-1 test facility at Corvallis and, later, posted Draft RAI 9380 Supplement.pdf, "Draft RAI 9380 Supplement" (Document 126) and Updated Draft RAI 9380 Supplement_update.pdf, "Updated Draft RAI 9380 Supplement Update" (Document 127), "2-14-19 Discussion.pdf" (Document 121) was put on the eRR to facilitate the discussion. Documents 126 and 127 provided comparison plots of CNV pressure, liquid temperature, liquid level, and RPV pressure on same respective graph for the 3 nodalization schemes versus time for the transient (up to 60 seconds past the containment peak pressure), and a table summarizing the nodalization sensitivity study results. The staff audit of the information showed that NuScale had performed containment NPM nodalization studies for the limiting peak CNV pressure DBE (inadvertent opening of an RRV) by evaluating the effects of using a set of coarser [redacted] and finer [redacted] axial nodalization of the containment volume and liquid space compared to the base case with [redacted] NuScale also performed sensitivity studies to evaluate the impact of using a finer radial nodalization of the containment wall [redacted] and a finer reactor pool axial nodalization [redacted] to evaluate the significance of their nodal representation with respect to containment peak pressure and temperature. The audited

information on NPM nodalization results showed that sufficiently fine nodalization had been used in the licensing-basis safety analysis, specifically in the liquid region and CNV axial volume nodalization had demonstrated enough fidelity for thermal stratification and subcooling of liquid water in the lower containment. Besides, the NPM containment pressure and temperature responses were shown to be not very sensitive [redacted] to various axial and radial nodalization to make a significant impact on the peak CNV pressure margin. On staff's request, the NuScale submitted a summary of the audited nodalization studies as a supplemental response to RAI 9380, Question 06.02.01.01.A-5 that included all the staff identified content. Later, the NuScale submitted a summary of the audited nodalization information as supplemental RAI response to resolve the staff concerns about NRELAP5's predictive capability for liquid thermal stratification and subcooling inside the containment, which was an open item in Phase 2 SER.

The staff also requested and NuScale provided similar containment nodalization studies for the NIST-1 facility containment for the HP-49 test conditions, through an audit document 9380_q6_supplement_draft_2019.04.22.pdf, "Draft Supplement to RAI 9380-4-22-19" (Document 128). That is a viable approach as HP-49 test conducted at the NIST-1 facility was the largest RPV liquid discharge into the containment at the lowest elevation that is why the staff expected HP-49 test to be most representative of the containment thermal-hydraulics phenomena including thermal stratification and subcooling involved in the limiting peak containment pressure DBE of an inadvertent RRV opening. The draft supplement of RAI 9380, Question 06.02.01.01.A-6 (Document 128) described how NRELAP5 calculates temperature stratification and presents peak pressure and collapsed liquid level results of an HP-49 re-nodalization study. The staff found insignificant variations in the predicted pressure response and the containment collapsed liquid level across three axial CNV nodalization. The peak containment pressure increased by 0.3 percent by changing from Base case [redacted] to finer [redacted] nodalization and [redacted] by changing from the Base case to coarser [redacted] nodalization, which indicates that the sensitivity of peak containment pressure to nodalization is not significant, considering the [redacted] margin available in the NPM design pressure. The staff concluded that the NIST-1 nodalization studies results and convergence trends are similar to that of the NPM nodalization studies, with finer nodalization leading to higher CNV pressure than the coarser nodalization. The finer and baseline predictions are closer, while coarse nodalization is less conservative. NuScale submitted a summary of the audited nodalization study of the NIST-1 facility for HP-49 test as a supplemental response to RAI 9380, Question 06.02.01.01.A-6 that included all of the staff identified content.

As the NPM and NIST-1 HP-49 nodalization results are consistent despite the scaling differences between the NPM and NIST-1 test facility, the staff did not identify any NRELAP5 bias around containment pressure or temperature predictions due to thermal stratification and subcooling of the liquid water in the lower containment. The audited documents showed that the applicant successfully demonstrated NRELAP5's capability to predict the peak containment pressure for both NPM and NIST-1 HP-49 test configuration, and no significant thermal-hydraulic phenomena were missed in the NRELAP model.

VI.B Containment Isolation

For this audit activity, the staff reviewed documents 16, 18, 54, 91, 109, and 110 listed in the Enclosure related to containment isolation valves. These documents are associated with NuScale's failure modes and effects analysis (FMEA) for the containment isolation valves and NuScale's approach to developing the environmental conditions applicable to the area where the containment isolation valves are located.

NuScale's FMEA is a single failure evaluation consistent with NuScale DSRS 6.2.4. The analysis is a system-level FMEA. The analysis describes the effect of failures on the containment isolation system and the fluid system associated with the failed component. For example, mechanical failure mechanisms include items such as wear, jamming, and damaged sealing surfaces. RAIs related to containment isolation valves and NuScale's FMEA (Document 91) are contained in eRAI 8820, Question 03.09.06-2 and eRAI 9075, Question 10.03-5. These RAIs note that NuScale's DCA Part 2, Tier 2 did not describe the FMEA for the containment isolation valves.

In the NuScale design, piping lines penetrating containment are situated in one general area, which is under the bioshield (also referred to the top of the module). These piping lines connect to the reactor coolant system (e.g., injection and discharge), steam & feedwater system, component cooling water system, containment flood and drain system, and the containment evacuation system. On these piping lines penetrating containment, NuScale's design includes containment isolation valves. All isolation valves are located outside containment and under the bioshield. Postulated line breaks outside containment and under the bioshield are evaluated in NuScale Part 2, Tier 2, Chapter 3 to determine the environmental conditions under the bioshield. Because all containment isolation valves are in the same general area (outside containment and under the bioshield), the environmental service conditions due to normal operations or a postulated pipe failure under the bioshield and its effect on the containment isolation safety function are important to evaluate due to the potential for the environment to lead to a common cause failure. NuScale documents (Documents 16, 18, and 109) describe a bioshield design and analysis that differs from the description provided in NuScale's Part 2, Tier 2, Chapter 3. The design of the bioshield (e.g., vertical face plate) directly impacts the development of the environmental conditions (e.g., pressure and temperature) under which SSC's important to safety are required to function. Due to the audit the staff gained insights into the complexity and significance of the bioshield design and associated analyses. As a result, the staff issued eRAI 9160, Question 03.02.01-4 through -10 to better understand the bioshield design basis.

After receiving the response to eRAI 9160, the staff asked a follow-up question, eRAI 9447, Question 3.11-19, requesting more information regarding the bioshield design. In response to the staff's questions, the applicant redesigned the bioshield. The redesign addressed venting of the bioshield, which directly impacts the development of environmental conditions under the bioshield. The modified design uses staggered panels creating a passive, open flow path. The staff audited the drawings (Document 54) and the associated calculation (Document 110). The calculation supporting the development of the environmental conditions under the bioshield appropriately modeled the vented bioshield (e.g., available flow area, hydraulic diameter, and loss coefficient) to include a stacked bioshield configuration (one bioshield affixed atop of the adjacent bioshield to support refueling activities).

Consistent with the regulatory basis for the audit (10 CFR 52.47 and GDC 54 through 57) the staff also generated RAIs that pertain to containment isolation provisions (eRAI 8863, Question 06.02.04-5) and testing of containment isolation provisions (eRAI 8863, Question 06.02.04-3). In addition to this audit activity, the staff also conducted a separate audit related to the containment isolation valve design. The audit results for that activity are provided in a separate report (ADAMS Accession No. ML18331A042).

VI.C Combustible Gas Control, including Exemption Request #2

To make a safety finding that the NuScale Combustible Gas Control design meets Title 10 *Code of Federal Regulations* (10 CFR) 50.44(c), as it relates to pressurized water-reactor (PWR) plants being designed to accommodate hydrogen generation equivalent to 100-percent fuel clad-coolant reaction while maintaining containment structural integrity, staff reviewed and relied on Documents 7, 9, 10, 57, and 65. This review included the structural impact in the containment vessel of both hydrogen combustion and detonation.

In part, 10 CFR 50.44(c) requires that all systems required to maintain containment integrity and safe shutdown must continue to perform their functions during DBAs and significant beyond-DBAs, including any effects resulting from the hydrogen generated following a fuel clad-coolant reaction involving 100 percent of the fuel cladding. To make a safety finding the staff reviewed and relied on the following Documents 55, 57, 58, 60, 62 during its review of the systems, structures and components which comprise the equipment survivability program. Since combustible gas control is primarily a significant beyond design basis review, the staff review, and conclusions are found in SER Sections 6.2.5 and 19.2.3.3.8. In Section 19.2.3.3.8 the staff reviewed the environmental conditions due to radiological consequences. The staff partially informed its review on Documents 76, 77, 79, and 80.

The staff evaluated NuScale's exemption request not to provide a means of controlling hydrogen concentration in containment to comply with 10 CFR 50.44(c)(2). This regulation requires limiting hydrogen concentrations following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 per cent fuel clad-coolant reaction to less than 10 per cent (by volume).

The staff evaluated the beyond-design-basis scenarios identified in FSAR Tier 2, Table 19.2-2, "Core Damage Simulations for Severe Accident Evaluation," which could result in generating and releasing hydrogen into an intact containment and concluded that these scenarios do not have enough oxygen to support combustion. The staff relied on Documents 70, 72, 74, 76, 77, 79, 81, 82, and 83.

VI.D Containment Leak Rate Testing

In order to evaluate the Exemption Request #7, for no design provisions for containment leakage Type A testing, staff reviewed Document 24, which: analyzed the cross-sectional area and thread engagement length of the CNV flanged opening bolts; calculated the required flange bolt preload values, in order to prevent leakage at the bolted flanges; and also analyzed the contact at the inner sealing surface for all containment bolted connections. This analysis included the containment conditions during the preservice design pressure leakage test, during an accident, and following lift conditions. This analysis and results were fundamental to staff's understanding and acceptance of the exemption request.

The staff review benefitted from module refueling operations in Document 92.

VI.E High Energy Line Break Outside Containment

The staff audited documents 103 through 108 listed in the enclosure. NuScale calculations show that SSCs are compatible with environmental conditions documented in the applicant's HELB calculations, which encompass the required event spectrum, including normal operations and postulated accidents. NuScale used appropriate input for its GOTHIC HELB model. The staff finds that NuScale calculations are reasonable.

V. CONCLUSION

The agreement between the NuScale licensing basis calculations of peak containment pressure and temperature and the MELCOR confirmatory analysis results is reasonable. Independent confirmatory calculations using MELCOR and TRACE codes show that NuScale containment peak pressure and temperature predicted by applicant using NRELA5 are conservative, and NuScale containment design has enough margin.

In the NuScale design, piping lines penetrating containment are situated in one general area, which is under the bioshield. The staff audited the documents on bioshield design and associated analyses. The audited documents supported the staff in issuing RAIs related to containment isolation provisions and testing.

The documents audited supported the staff review of NuScale combustible gas control design to determine that it meets 10 CFR 50.44(c), as it relates to PWR plants being designed to accommodate hydrogen generation equivalent to 100-percent fuel clad-coolant reaction while maintaining containment structural integrity, and related Exemption Request #2 to not provide a means of controlling hydrogen concentration in containment to comply with 10 CFR 50.44(c)(2).

NuScale analysis provided for the audit included the containment conditions during the preservice design pressure leakage test, during an accident, and following lift conditions, which enhanced the staff understanding of Exemption Request #7 on containment leak rate testing.

NuScale calculations on high energy line break outside containment show that SSCs are compatible with environmental conditions documented in the applicant's HELB calculations, which encompass the required event spectrum, including normal operations and postulated accidents. NuScale used appropriate input for its GOTHIC HELB model. The staff finds that NuScale calculations are reasonable.

VI REFERENCES

1 “Containment Response Analysis Methodology” Technical Report, TR-0516-49084-P, Revision 0 (ADAMS Accession No. ML17009A491), January 2017, NuScale Power, LLC.

2 “Loss-of-Coolant Accident Evaluation Model" Topical Report, TR-0516-49422-P, Revision 0, Proprietary version, December 2016, NuScale Power, LLC.

3 “Non-LOCA Transient Analysis Methodology” Topical Report, TR-0516-49416-P, Revision 0, proprietary version, January 2017, NuScale Power, LLC.

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8 Colburn, A.P., and O.A. Hougen, "Design of Cooler Condensers for Mixtures of Vapors with Noncondensing Gases," Industrial and Engineering Chemistry, (1934): 26:1178-1182.

9 Churchill, S.W., and Chu, H.H.S., 1975. “Correlating Equations for Laminar and Turbulent Free Convection from a Vertical Plate,” International Journal of Heat and Mass Transfer, Vol. 18, pp. 1323-1329.

10 McAdams, W.H., “Heat Transmission,” 3rd Edition, McGraw-Hill, New York, NY, 1954.

Attachment: Documents Audited by the Staff

No.	File	Document Name	Rev. #
1	00002090_1_RPV and CNV Flange Geometry Stud.pdf	RPV and CNV Flange Geometry	1
2	32-9257575-000 NuScale Reactor Core Chemical Deposition Analysis.pdf	NuScale Reactor Core Chemical Deposition Analysis	
3	51-9257323-000 AIS for NuScale GSI-191 Evaluation.pdf	AIS for NuScale GSI-191 Evaluation	
4	DD-F010-4444_R0_Bioshield_Re-Design_to_Support_Environmental_Qualification_Profile.pdf	Bioshield Re-Design to Support Environmental Qualification Profile	0
5	DI-0303-51058_R0.pdf	NIST-1 Data Processing for Code Assessment Purposes	0
6	EC_0000_3853_R1_Calcs_to_Support_NIST-1_Distortion_Analysis_and_Modeling_of_Containment_and_Pool_Heat_Transfer	Calculations to Support NIST-1 Distortion Analysis and Modeling of Containment and Pool Heat Transfer	1
7	EC_A013_00003036_01_CNV_Ultimate_Pressure_Integrity_Analysis.pdf	CNV Ultimate Pressure Integrity Analysis	1
8	EC_A013_3377_R0_CNV_Primary_Stress_Analysis.pdf	Primary Stress Analysis of the Containment Vessel	0
9	EC_B020_2877_R1 ECCS Combustible Gas Analysis.pdf	ECCS Combustible Gas Analysis	1
10	EC_B020_4365_R0, Containment Gas Composition Calculation.pdf	Containment Gas Composition Calculation	0
11	EC_F010_5233_00_CNV_Support_Interface_with_RXB_Floor.pdf	CNV Interface with RXB Floor	0
12	EC-0000-2250-R0_Feedwater_Piping_Failure_Analysis.pdf	Feedwater Piping Failure Analysis	0
13	EC-0000-2714-R0_Steam_System_Piping_Failure_Analysis.pdf	Steam System Piping Failure Analysis	0
14	EC-0000-2786_R2_Failure_of_Small_Lines_Carrying_Primary_Coolant_Outside_Containment.pdf	Failure of Small Lines Carrying Primary Coolant Outside Containment	2
15	EC-0000-4718_R0_GOTHIC_HELB_Cases_of_NS_Top_of _Model_and_RXB_Pool_Room.pdf	GOTHIC HELB Cases of NuScale Top of Module RXB Pool Room	0

No.	File	Document Name	Rev. #
16	EC-0000-4720_R1_NS_High_Energy_Line_Break_Scenario_Definition_Top_of_Module.pdf	NuScale High Energy Line Break Scenario Definition - Top of Module	1
17	EC-0000-4721 1_GOTHIC_HELB_Causes_of_NS_Top_of_Module_and_RXB_Pool_Room	GOTHIC HELB Cases of NuScale Top of Module RXB Pool Room	1
18	EC-0000-4745_R0_GOTHIC_HELIB_Cases_of_NS_Top_of_Module_and RXB_Pool_Room_BioShield_Blowout_Panels.pdf	GOTHIC HELB Cases of NuScale Top of Module and RXB Pool Room with Bioshield Blowout Panels Design	0
19	EC-0000-4746_R0_GOTHIC_HELIB_Cases_of_Revised_M&E_for_NS_Top_of_Module_and_RXB_Pool_Room.pdf	GOTHIC HELB Cases of Revised M&E for NuScale Top of Module and RXB Pool Room	0
20	EC-0000-5435-R0_final.pdf	Containment Pressure Initial Condition Sensitivity Calculations	0
21	EC-A010-2322-R3 Reactor Module Seismic Model.pdf	Reactor Module Seismic Model	3
22	EC-A010-3559-R3 Reactor Module Seismic Calculation.pdf	Reactor Module Seismic Calculation	3
23	EC-A010-4270-1.pdf	Long Term Cooling Analysis	1
24	EC-A013-1691_R0 Containment Vessel Flange Bolting Calculation.pdf	Containment Vessel Flange Bolting Calculation	0
25	EC-A013-2341 2 Contain Pressure Temperature Response Design Basis Events Analysis.pdf	Containment Pressure and Temperature Response to Design Basis Events	2
26	EC-A013-2341_R1 Containment Pressure and Temperature Response Design Basis Events Analysis.pdf	Containment Pressure and Temperature Response to Design Basis Events	1
27	EC-A030-4101, R0 Class 1 Piping Stress Analysis for RCS Discharge_Line.pdf	Class 1 Piping Stress Analysis for RCS Discharge Line	0
28	EC-B060-4543 0.pdf	GOTHIC Passive Cooling of NuScale Control Room Building	1
29	EC-B060-4544_ R0_GOTHIC_Passive_Cooling_Analysis_of_NS_RXB_Main_Pool_Room.pdf	GOTHIC Passive Cooling Analysis of NuScale RXB Main Pool Room	1
30	EC-B175-3253_R0 Ultimate Heat Sink Boil Off Calculation.pdf	Ultimate Heat Sink Boil Off Calculation	0
31	EC-F010-3108 Rev. 7.pdf	Seismic Soil-Structure Interaction Analysis of NuScale Reactor Building for ISRS Generation	0

32	ECN_A010_4189_R0_For EQ-A010-3642 Correction to Table 3-1.pdf	Corrections to Table 3-1 Normal Operating Pressures and Temperatures for DHRS Lines (for EQ-A010-3642, 0)	0
33	ECN_A010_4575_R0 for EQ-A010-3642 Overpress Protect Require Steam Gen System Piping.pdf	Revision of Overpressure Protection Requirements for Steam Generator System Piping (for EQ-A010-3642, 0)	0
34	ECN_A010_4614_R0 for EQ-A010-2224 2nd Systems Contain Isolate Valves Design Spec Term Change.pdf	Secondary Systems Containment Isolation Valves Design Spec. Terminology Change (for EQ-A010-2224, 0)	0
35	ECN_A010_4981_R0 for EQ-A010-3642, RXM Class 1, 2, 3 Piping Design Spec. Terminology Change.pdf	RXM Class 1, 2, 3 Piping Design Spec. Terminology Change (for EQ-A010-3642, 0)	0
36	ECN_A010_5024_R0 for EQ-A010-2224 Material Specification Update.pdf	Material Specification Update (for EQ-A010-2224, 0)	0
37	ECN_B020_4991_R0 Add Evaluation of Hydrogen Mixing during ECCS operation.pdf	Add Evaluation of Hydrogen Mixing during ECCS Operation (for EC-BZ020-4365, 0)	0
38	ECN_B020_5023_R0 for EQ-B020-2140 Material Specification Update.pdf	Material Specification Update (for EQ-A010-2140, 2)	0
39	ECN_B030_4700_R0_For EQ-B030-2258 ASME Design Specs Decay Heat Removal System Actuation Valves.pdf	ASME Design Specification for Decay Heat Removal System Actuation Valves (for EQ-B030-2258, 0)	0
40	ECN_B030_4849_R0_For EQ-B030-2258 Decay Heat Remove Actuation Valves Load Combine Changes.pdf	Decay Heat Removal Actuation Valves Design Specification Terminology and Load Combination Changes (for EQ-B030-2258, 0)	0
41	ECN_B090_4290_Deisgn Solution for the HELB FR.pdf	Design Solution for the HELB FR (for SD-B090-1680, 0)	0
42	ECN_B090_4436_Add Passive Vent requirement to the FS.pdf	Add Passive Vent Requirements to the FS (for FS-B090-0533, 3)	0
43	ECN-0000_4908_R0 Containment Pressure and Temperature Updates ER-0000-4316.pdf	Containment Pressure and Temperature Updates (for ER-0000-4908, 1)	0
44	ECN-0000-4968_R0 Update Appendix A Table A-1 Environmental Zones ER-0000-4316.pdf	Update Appendix A Table A-1, Environmental Zones Required Update (for ER-0000-4316, 1)	0
45	ECN-0000-4998_R0 Various Updates to ER-0000-4316 Body, Figures and Tables.pdf	Various Updates to ER-0000-4316 Body, Figures and Tables (for ER-0000-436, 1)	0
46	ECN-0000-5032_R0 for ER-0000-3921 Reorganization of Chapter 5.0.pdf	Reorganization of Chapter 5.0 (for ER-0000-3921, 0)	

47	ECN-0000-5033_R0 for ER-0000-3921 Misc. Wording Adjustments Licensing Purposes.pdf	Miscellaneous Wording Adjustments for Licensing Purposes (for ER-0000-3921, 0)	0
48	ECN-0000-5036_R0 for ER-000-3921 Reorganization of Chapter 3.0.pdf	Reorganization of Chapter 3.0 (for ER-0000-3921, 0)	0
49	ECN-A010-4742_R0 for EQ-A010-2224 ASME Design Specs Second Systems Contain Isolation Valves.pdf	ECN for ASME Design Specification for Secondary Systems Containment Isolation Valves (for EQ-A010-2224, 0)	0
50	ECN-A013-5079_R0 for EC-A013-2341 Additional M&E Tables for Licensing.pdf	ECN for EC-A013-2341, 2 (Containment Pressure and Temperature Response to Design Basis Events)	
51	ECN-A013-5131_R2 for EC-A013-2341 O-RELAP v1.3.0 input decks for mass unit conversion.pdf	Adding M&E Tables for Licensing (for EC-A013-2341, 2)	0
52	ECN-B030-4744_R0 for EQ-B030-2258 ASME Design Specs Decay Heat Remove System Actuation Valves.pdf	ECN for ASME Design Specification for Decay Heat Removal System Actuation Valves (for EQ-B030-2258, 0)	0
53	EC-T080-3822-R1.pdf	NRELAP5 Assessment Against NuScale Separate Effects High Pressure Condensation Test Series NIST-1 HP-02	1
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57	EQ_A013_00001826_01_ASME Design Specification for Containment_Vessel.pdf	ASME Design Specification for Containment Vessel	1
58	EQ_B020_00002140_02_ASME_Design_Specification_for_Emergency_Core_Cooling_Valvespdf	ASME Design Specification for Emergency Core Cooling System Valves	2
59	EQ-A010-2224_R0 ASME Design Specification for Secondary System Containment Isolation Valves.pdf	ASME Design Specifications for Secondary Systems Containment Isolation Valves	0
60	EQ-A010-2235_R0 ASME Design Specification for Primary Systems Containment Isolation Valves.pdf	ASME Design Specifications for Primary Systems Containment Isolation Valves	0
61	EQ-A010-3642_R0 ASME Design Specification for RXM Class 1 2 3 Piping.pdf	ASME Design Specifications for RXM Class 1, 2, and 3 Piping	0

62	EQ-A013-5418 R0 - ASME Design Specification for Containment EPAs.pdf	Design Specification for CNV Electrical Penetration Assemblies	0
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70	ER_P020_00004896_00_Severe_Accident_Selection_Methodology.pdf	Severe Accident Selection Methodology	0
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72	ER_P020_7024_R0_Level_2_Probabilistic_Risk_Assessment_Notebook_wECN	Level 2 Probabilistic Risk Assessment Notebook	0
73	ER_P060_00007077_A_TRN_16T__General_Transient_with_Two_Trains_of _DHRS__Reactor_Recirculation_Valve _Open__Loss_of_DC_Power_.pdf	TRN-16T: General Transient with Two Trains of DHRS and Reactor Recirculation Valves Open (Loss of DC Power)	A
74	ER_P060_4715_R0_TRN_07T__General_Transient_with_Stuck_Open_RSV_ and_No_Mitigation.pdf	TRN-07: General Transient with Stuck Open Reactor Safety Valve and No Mitigation, from a PRA Level 2 Perspective	0
75	ER_P060_4724_R0 NuScale MELCOR Basemodel.pdf	NuScale MELCOR Basemodel	0
76	ER_P060_4748_R0_LEC_06T__RVV_ LOCA_with_No_Mitigation.pdf	LEC-06T: Reactor Vent Valve LOCA with No Mitigation, from a PRA Level 2 Perspective	0
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79	ER_P060_4857_R0_LCC_05T__Charging_Line_Break_Inside_Cntmt_Complete_ECCS_Failure_wECN.pdf	LCC-05T: Charging Line Break Inside Containment with Complete ECCS Failure, from a PRA Level 2 Perspective	0
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84	ER_P060_7082_R0_NRELAP5_PRA_Base_Model.pdf	NuScale NRELAP5 Probabilistic Risk Assessment Base Model	0
85	ER-0000-2486 4 Safety Analysis Analytical Limits report.pdf	Safety Analysis Analytical Limits Report	4
86	ER-0000-2486 Revision 5.pdf	Safety Analysis Analytical Limits Report	5
87	ER-0000-3921_R0 Long Term Core Cooling Methodology Report.pdf	Long Term Core Cooling Methodology Report	0
88	ER-0000-4316_R1 Environ Service Conditions Electrical and Mechanical Equipment Qualification.pdf	Environmental Service Conditions for Electrical and Mechanical Equipment Qualifications	1
89	ER-0000-4391 0_Mass_and_Energy_Release_and_Containment_Vessel_Pressure_and_Temp_Response_Method	Mass and Energy Release and Containment Vessel Pressure and Temperature Response Methodology	0
90	ER-A013-3246_R0 Containment Vessel Structural Eval for Combustible Gas.pdf	Containment Vessel Structural Evaluation for Combustible Gas	0
91	ER-A013-3635_R0 Containment System Failure Modes and Effects Analysis.pdf	Containment System Failure Modes and Effects Analysis	0
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104	TR EC-0000-4718	GOTHIC HELB Model of NuScale TOP of Module and RXB Pool Room	0
105	TR EC-0000-4720	NuScale High Energy Line Break Scenario Definition – Top of Module	1
106	TR EC-0000-4721	GOTHIC FELB Cases of NuScale Top of Model and RXB Pool Room	1
107	TR EC-0000-4745	GOTHIC HELB Cases of NuScale TOP of Model and RXB Pool Room with Bioshield Blowout Panels Design	0
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109	ER-0000-4316	Environmental Service Conditions for Electrical and Mechanical Equipment Qualification	1
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112	EC-A013-2341_R3 with ECN.pdf	Containment Pressure and Temperature Response to Design Basis Events Analysis - with ECN	3

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