NUSCALE POWER™

RAIO-0718-60761

July 03, 2018

Docket: PROJ0769

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 9466 (eRAI No. 9466) on the NuScale Topical Report, "Non-Loss of Coolant Accident Analysis Methodology," TR-0516-49416, Revision 1

REFERENCES:

1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 9466 (eRAI No. 9466)," dated May 07, 2018
2. NuScale Topical Report, "Non-Loss of Coolant Accident Analysis Methodology," TR-0516-49416, Revision 1, dated August 2017

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 Questions from NRC eRAI No. 9466:

- 15.00.02-7
- 15.00.02-8
- 15.00.02-13

The response schedule for the remaining questions of RAI No. 9466, eRAI No. 9466 were provided in an email to NRC (Greg Cranston) dated June 19, 2018.

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 9466 (eRAI No. 9466). 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
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Rani Franovich, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9466, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 9466, nonproprietary

Enclosure 3: Affidavit of Zackary W. Rad, AF-0718-60761

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Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9466, proprietary

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 9466, nonproprietary

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

eRAI No.: 9466
Date of RAI Issue: 05/07/2018

NRC Question No.: 15.00.02-7

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. In addition, GDC 15 requires that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.

TR-0516-49416-P supports the conclusions relative to GDC 10 and 15 in the NuScale FSAR, which under 10 CFR 52.47 must describe the facility, present the design bases and the limits on its operation, and present a safety analysis of the structures, systems, and components and of the facility as a whole. SRP Section 15.0.2 provides the staff guidance on reviewing analytical models and computer codes used to analyze transient and accident behavior. SRP Section states:

Models must be present for all phenomena and components that have been determined to be important or necessary to simulate the accident under consideration. The chosen mathematical models and the numerical solution of those models must be able to predict the important physical phenomena reasonably well from both qualitative and quantitative points of view.

TR Section 6.1.2 discusses the core kinetics in the NRELAP5 plant model of the NuScale power module (NPM) and states:

The fission product decay type is specified as 'gamma-ac' with the 'ans73' model, which calculates decay heat in accordance with the 1973 ANS standard while adding the contribution from actinides. A fission product yield factor of 1.0 is specified in the base model, which can be changed to suit the scenario being analyzed.

According to TR Section 7.1.5.3, decay heat is biased either low or high, depending on the transient being analyzed, by use of decay heat multipliers and specifying whether or not to include the actinide contribution. The staff requires justification that, as altered by multipliers and actinide contribution in the non-LOCA analyses, the 1973 model leads to conservative results. In response to an audit discussion (Round 2, Issue 5), the applicant stated in the quality assurance process for NRELAP5, some of the decay heat models were subjected to greater scrutiny, with the 1973 decay heat model being one of greater pedigree. Therefore, it was selected for use. The applicant stated that a calculation that examined the various decay heat models was performed, and the results helped determine that a multiplier for a maximum or minimum decay heat level was appropriate.

Furthermore, Page 112 of the LOCA submittal (TR-0516-49422-P) states that "λ and η values can be user-specified, or default values equal to those stated in the 1979 ANS standard (Table 6-4), the 1994 Standard, or the 2005 Standard can be used."

Because decay heat affects energy production in the core and therefore transient progression, the staff requires additional information about the use of the 1973 decay heat model and the methodology for selecting the values used for λ and η.

Information Requested:

Provide the actual values of λ and η used and references for these parameters, and justify that the use of these values, the specified multipliers, and inclusion (or lack thereof) of actinides in combination with the 1973 decay heat model leads to a conservative result for decay heat contribution for all event types. Update TR-0516-49416-P and any other affected documentation as appropriate.

NuScale Response:

As discussed in Section 5.1.2.2.5 of the LOCA topical report (TR-0516-49422-P), the 'gamma-ac' option activates the models for the transient effects of decay heat and actinides. The specific models activated are the ANS73 model for decay heat and the ANS79 model for actinides.

The discussion in Section 6.10 of the LOCA topical report (TR-0516-49422-P) indicates that the default values for lambda (λ) and eta (η) are found in Table 6-4. These values are repeated below.

Table 6-4. ANS-79 actinide model constants.

Isotope	(s^{-1})	(MeV)
^{239}U	1.772	0.00299
^{239}Np	0.5774	0.00825

The limiting nature of a specific combination of decay heat curve, decay heat multiplier, and actinide contribution is confirmed each cycle as discussed in Section 7.1.5.3 of TR-0516-49416-P the Non-LOCA topical report.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

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

eRAI No.: 9466
Date of RAI Issue: 05/07/2018

NRC Question No.: 15.00.02-8

GDC 15 requires that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. TR-0516-49416-P supports the conclusions relative to GDC 15 in the NuScale FSAR, which under 10 CFR 52.47 must describe the facility, present the design bases and the limits on its operation, and present a safety analysis of the structures, systems, and components and of the facility as a whole.

RG 1.203 describes the evaluation model development and assessment process (EMDAP), which the NRC staff considers acceptable for use in developing and assessing EMs used to analyze transient and accident behavior. Step 18 of the EMDAP (Section 1.4.6) discusses preparation of input and performance of calculations to assess system interactions:

> *"The ability of the EM to model system interactions should also be evaluated in this step, and plant input decks should be prepared for the target applications. Sufficient analyses should be performed to determine parameter ranges expected in the nuclear power plant."*

TR-0516-49416-P, Table 7-45 provides the results of representative sensitivity studies for the loss of normal AC power event. All peak primary pressures in Table 7-45 are {{

}}$^{2(a),(c)}$.

In addition, Table 7-45 shows that the parameter with the largest effect on the peak SG pressure is {{

}}$^{2(a),(c)}$.

The staff observes that reducing the {{

}}2(a),(c).

The staff needs additional information to understand these sensitivity studies since similar sensitivity studies are conducted for the FSAR analyses to identify the initial conditions and biases that result in the most limiting figures of merit.

Information Requested:

a. Clarify the timing and delay times for the reactor trip, decay heat removal system (DHRS) activation time, DHRS valve time to complete stroke, the time to when AC power to the normal DC power system (EDNS) battery chargers is lost, and the time at which the containment is fully isolated.
b. Primary side peak pressure is {{

}}2(a),(c) would result in a higher primary side peak pressure.
c. Explain the reductions in the peak SG pressure for {{

}}2(a),(c).
d. Explain the relatively large variation in the peak SG pressure for {{

}}2(a),(c).

NuScale Response:

<u>Response to part a) of request:</u>

Table 1 provides the timing for detecting and actuating the reactor trip system (RTS), the decay heat removal system (DHRS), and the containment (CNV) isolation system for the loss of normal AC power cases listed in TR-0516-49416-P Table 7-45.

The main steam isolation valves (MSIVs) and the feedwater isolation valves (FWIVs) have a closing stroke time of 5 seconds. The secondary MSIVs (SMSIVs) and the feedwater regulating valves (FWRVs) have a closing stroke time of 30 seconds. The DHRS actuation valves have an opening stroke time of 30 seconds.

AC power is assumed to be lost to the normal DC power system (EDNS) batteries at event initiation.

The containment is isolated once the containment isolation valves are fully closed using the valve-specific closing stroke times following generation of the containment isolation signal as listed in Table 1. Note that some valves may be closed sooner based on generation of the DHRS actuation signal.

Table 1 - Times to reach Analytical Limits and actuate RTS, DHRS, and CNV isolation for loss of normal AC power cases in TR-0516-49416-P Table 7-45

Case Number	Analytical Limit for RTS & DHRS Reached, sec	RTS & DHRS Signal Generated, sec	Analytical Limit for CNV Isolation Reached, sec	CNV Isolation Signal Generated, sec
1	5.36	7.37	20.16	22.16
2	5.36	7.36	19.72	21.72
3	5.52	7.52	19.63	21.63
4	8.70	10.7	23.92	25.92
5	5.47	7.48	21.60	23.60
6	5.49	7.50	19.08	21.08
7	8.40	10.40	23.60	25.60
8	8.58	10.58	24.02	26.02
9	5.41	7.42	20.79	22.79
10	5.52	7.53	20.49	22.49
11	8.64	10.64	24.44	26.44
12	8.43	10.44	23.03	25.03
13	5.57	7.57	19.94	21.94
14	8.80	10.80	23.54	25.54
15	8.57	10.57	24.18	26.18
16	5.50	7.51	18.62	20.62
17	9.12	11.12	60.0	62.0
18	10.75	12.75	33.80	35.80
19	10.39	12.40	39.41	41.41
20	5.36	7.37	20.16	22.16

Response to part b) of request:

Yes, the general invariance to peak reactor pressure vessel (RPV) pressure is due to the relief capacity of the reactor safety valve (RSV).

The peak RPV pressure for Case 17 is lower than that for Case 19 because reactor trip and DHRS actuation occur sooner for Case 17 than for Case 19 (see Table 1 of the response to part a) of this request). The difference in trip time works in conjunction with the initial biasing to keep the pressurizer pressure from reaching the actuation pressure (2137.25 psia) for RSV-1. It is also worth noting that RSV-1 did not lift for the nominal biasing case (peak RPV pressure of 2094 psia).

Response to part c) of request:

When comparing for initial T_{avg} bias: Case 3 (high, +10°F) to Case 6 (nominal) to Case 16 (low, -10°F), the sensitivity of peak SG pressure to RCS temperature is because DHRS operates in a boiling condensing mode. Thus, a higher initial RCS temperature translates to a higher Psat in the DHRS loop. Since RCS temperature is a significant driver for peak SG pressure the initial T_{avg} has a larger effect than changes in the initial SG pressure.

Response to part d) of request:

As described in the response to part c, the high biased RCS temperature is a primary driver of the peak secondary pressure but this effect is not linear as shown in cases 3, 6 and 16. Additionally, high steam pressure bias will tend to maximize SG pressure because of the increased initial SG inventory as shown in the cited sensitivity results (case 1 vs 5, 2 vs 9, 3 vs 10, 4 vs 11, 6 vs 13, 8 vs 14 and 7 vs 15). Consistent with this description, cases 1 vs 5 have a 60 psi difference while cases 2 and 9 only have a difference of 34 psi indicating that increased initial SG inventory is consistently more limiting but is even further exacerbating when combined with high biased RCS temperature. RCS pressure is actually quite insensitive as demonstrated by cases 1, 4 and 17 which have a total variation of 4 psi when the other more important parameters of RCS temperature and initial SG pressure are biased high.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9466
Date of RAI Issue: 05/07/2018

NRC Question No.: 15.00.02-13

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. In addition, GDC 15 requires that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.

TR-0516-49416-P supports the conclusions relative to GDC 10 and 15 in the NuScale FSAR, which under 10 CFR 52.47 must describe the facility, present the design bases and the limits on its operation, and present a safety analysis of the structures, systems, and components and of the facility as a whole. SRP Section 15.0.2 directs the staff to review analytical models and computer codes used to analyze transient and accident behavior and states that a theory manual that describes such items as field equations, closure relationships, numerical solution techniques, etc. should be included as part of the EM documentation.

The staff notes that TR-0516-49416-P refers to the NRELAP5 theory manual but does not include it as a reference in TR Section 11.0. The adequacy of NRELAP5 and its ability to calculate pertinent physical phenomena is dependent on the physical and numerical modeling described in the theory manual. Therefore, please add the NRELAP5 theory manual as a reference in TR-0516-49416-P.

NuScale Response:

The reference for the NRELAP5 theory manual was added to the Non-LOCA Methodology topical report (TR-0716-50350) as Reference 25. Reference was made to it as indicated in the markup below.

Impact on Topical Report:

Topical Report TR-0516-49416, Non-Loss of Coolant Accident Analysis Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

pressure increases. {{

}}2(a),(c)

In the NuScale design, for a given reactor module operating condition, reactor power, core inlet temperature and system flow rate are tightly coupled. As described in Section 7.1, ranges in these parameters are considered as part of biasing the system transient analysis steady state initial conditions. The NRELAP5 system analysis methodology for determining the limiting CHF cases for downstream subchannel analysis is primarily dependent on the limiting initialization. The CHF cases are evaluated at the minimum flow initialization. Other initial conditions are forced to the limiting initialization for a given transient progression to ensure the maximum power, primary pressure and core inlet fluid temperature are simultaneously reached prior to reactor trip system actuation. For example, in the case of a heatup event, the RCS will increase in temperature, causing a pressurizer insurge and subsequent increase in pressure. The limiting CHF scenario is the transient progression that results in the highest core outlet temperature at the time of reactor trip on high pressure, which is generally the faster heatups where the pressurizer initialization is biased to delay the high pressure trip.

For some transients, a spectrum of cases is analyzed from the limiting initialization. {{

}}2(a)(c)

After the system transient analysis calculations are performed and assessed, for events that require subchannel analysis, one or more cases are identified as limiting for

15. U.S. Nuclear Regulatory Commission, “Standard Review Plan, Transient and Accident Analyses,” NUREG-0800, Chapter 15, Revision 3, March 2007 and various Chapter 15 subsections.

16. *U.S. Code of Federal Regulations*, “Definitions,” Section 50.2, Part 50 Chapter I, Title 10, “Energy,” (10 CFR 50.2).

17. American National Standards Institute / American Nuclear Society, “Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems,” ANSI/ANS-58.9-2002 R2015.

18. U.S. Nuclear Regulatory Commission, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,”SECY-94-084, March 1994.

19. *U.S. Code of Federal Regulations*, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” Appendix B, Part 50 Chapter I, Title 10, “Energy,” (10 CFR 50 Appendix B).

20. American Society of Mechanical Engineers, *Quality Assurance Program Requirements for Nuclear Facility Applications*, ASME NQA-1-2008, NQA-1a-2009 Addenda.

21. NuScale Power, LLC, “Rod Ejection Accident Methodology,” TR-0716-50350-P, Revision 0.

22. NuScale Power, LLC, “Applicability of AREVA Methodology for the NuScale Design,” TR-0116-20825-P, Revision 1.

23. NuScale Power, LLC, “Nuclear Analysis Codes and Methods Qualification,” TR-0616-48793-NP, Revision 0.

24. Kim, S.J., “Turbulent film condensation of high pressure steam in a vertical tube of passive secondary consensation system,” PhD thesis, Korea Advanced Institute of Science and Technology, 2000.

25. SwUM-0304-17023, Revision 5, NRELAP5 Version 1.3 Theory Manual.

Enclosure 3:

Affidavit of Zackary W. Rad, AF-0718-60761

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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 non-loss of coolant accident analysis methodology .

   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. 9466, eRAI No. 9466. 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 July 3, 2018.

Zackary W. Rad