UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 29, 2018

MEMORANDUM TO: Samuel S. Lee, Chief
Licensing Branch 1
Division of New Reactor Licensing
Office of New Reactors

FROM: Marieliz Vera, Project Manager **/RA/**
Licensing Branch 1
Division of New Reactor Licensing
Office of New Reactors

SUBJECT: SUMMARY OF THE FEBRUARY 14, 2018, CATEGORY 1 PUBLIC TELECONFERENCE WITH NUSCALE POWER, LLC DESIGN CERTIFICATION APPLICATION REQUEST FOR ADDITIONAL INFORMATION NOS. 8963 AND 8901 RESPONSES

The U.S. Nuclear Regulatory Commission (NRC) held a Category 1 public teleconference on February 14, 2018, to discuss the responses to NuScale Power, LLC (NuScale) Design Certification, Requests for Additional Information (RAI) Nos. 8963 and 8901, regarding Final Safety Analysis Report Tier 2, Chapter 3, "Design of Structures, Systems, Components and Equipment." Participants included personnel from NuScale and no members of the public.

The public meeting notice can be found in the Agencywide Documents Access and Management Systems under Accession No. ML18037A904. This meeting notice was also posted on the NRC public Website.

The meeting agenda and list of participants can be found in Enclosures 1 and 2, respectively. Some of the technical issues discussed are included in Enclosure 3.

CONTACT: Marieliz Vera, NRO/DNRL
301-415-5861

Summary:

The purpose of this meeting was to discuss the responses for RAIs 8963 Questions 03.08.05-7, -12, -13, -14 and -22 (ML17290B267) and RAI 8901 (ML17284A092). The NRC staff discussed their feedback to the responses (Enclosure 3) with NuScale. For RAI 8963, NuScale will address the NRC staff feedback by supplementing the RAI response. For RAI 8901, NuScale clarified portions of the feedback, and because of these clarifications the NRC staff will revise the feedback and have a follow-up public call.

Docket No. 52-048

Enclosures:
1. Meeting Agenda
2. List of Attendees
3. Comments presented by NRC staff

cc w/encls.: DC NuScale Power, LLC Listserv

SUBJECT: SUMMARY OF THE JANUARY 30, 2018, CATEGORY 1 PUBLIC TELECONFERENCE WITH NUSCALE POWER, LLC DESIGN CERTIFICATION APPLICATION REQUEST FOR ADDITIONAL INFORMATION NOS. 8963 AND 8901 RESPONSES DATE: 3/29/2018

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CATEGORY 1 PUBLIC TELECONFERENCE WITH NUSCALE POWER, LLC

DESIGN CERTIFICATION APPLICATION RESPONSES TO REQUEST FOR

ADDITIONAL INFORMATION NOS. 8963 AND 8901

February 14, 2018

1:00 p.m. – 3:00 p.m.

AGENDA

Public Meeting	
1:00-1:05pm	Welcome and Introductions
1:05-1:55pm	Discussion of the Request for Additional Information (RAI) 8963
1:55-2:00pm	Public - Questions and Comments
2:00-2:55pm	Discussion of the RAI 8901
2:55-3:00pm	Public - Questions and Comments

U.S. NUCLEAR REGULATORY COMMISSION

CATEGORY 1 PUBLIC TELECONFERENCE WITH NUSCALE POWER, LLC

DESIGN CERTIFICATION APPLICATION RESPONSES TO REQUEST FOR

ADDITIONAL INFORMATION NOS. 8963 AND 8901

LIST OF ATTENDEES

February 14, 2018

NAME	AFFILIATION
Marieliz Vera	U.S. Nuclear regulatory Commission (NRC)
Yiu Law	NRC
Timothy Lupold	NRC
Ata Istar	NRC
Marty Bryan	NuScale Power, LLC (NuScale)
Josh Parker	NuScale
Evren Ulku	NuScale
Nicholas Brown	NuScale
Giulio Flores	NuScale
Craig Harwood	NuScale
Tom Ryan	NuScale
Kyra Perkins	NuScale
Mohsin Khan	NuScale

U.S. NUCLEAR REGULATORY COMMISSION

CATEGORY 1 PUBLIC TELECONFERENCE WITH NUSCALE POWER, LLC

DESIGN CERTIFICATION APPLICATION RESPONSES TO REQUEST FOR

ADDITIONAL INFORMATION NOS. 8963 AND 8901

Staff feedback on Request Additional for Information 8963

Question 03.08.05-7

In its response, the applicant provided the basis for the surcharge pressure of 0.25 kilopounds per square foot (ksf) as that the most critical of either a minimum surcharge outside and adjacent to a subsurface wall due to wheel load converted to lateral equivalent or a railroad surcharge.

However, it appears that the applicant did not consider the potential differences in surcharge pressures on the walls of Reactor Building (RXB) and Control Building (CRB) due to the effects of adjacent buildings. Therefore, the applicant is requested to describe why the differences in surcharge pressures due to the effects of adjacent buildings were not determined/considered.

NOTES:

1. if there is a difference in surcharge pressures on the walls of RXB and CRB due to the effects of adjacent buildings that difference may need to be considered in the stability evaluations for the RXB and CRB also.

2. In deeply embedded structures, the surcharge-pressure loadings may have substantial contribution in the design of basemats.

Question 03.08.05-12

In its response the applicant provided two tables (Tables 1 and 2) describing “Building Models” and “Basemat Model” of RXB Basemat designs.

The applicant concluded that there is no impact to the final safety analysis report (FSAR) in their response. However, the applicant’s response provides information that should have been included (as markups) in Section 3.8.5.4.1.2, “RXB Basemat Model Description,” of the FSAR, to describe the models used for the RXB basemat designs. Furthermore, the NRC staff is requesting the applicant to respond the following questions:

1. It is not clear to the staff whether deleting and restraining elements 10 feet (ft.) above the RXB basemat would provide critical flexural stiffness that would provide conservative results for the design of the RXB basemat. Therefore, sensitivity evaluation(s) may need to be performed -- unless otherwise, restraining the elements 10 ft above the basemat can be justified to produce the worst boundary condition.

2. It is not clear to the staff how the external loads considered in the design of RXB basemat (e.g.; walls), since all structural components 10 ft. above the basemat were deleted.

3. Is the CRB basemat model constructed in a similar fashion? If so, the applicant is requested to describe the CRB basemat Model in detail.

Question 03.08.05-13

Based on the review of the applicant's response, it is not clear to the NRC staff how the reduction in 50 percent soil stiffness affects the design of basemat. Since, in this case, the settlements' of the basemats would be subjected to "rigid-body motions" only.

Question 03.08.05-14

Based on the review of the applicant's response, it is not clear to the NRC staff where/how the increase in basemat bending moments due to "stiff and soft spots" beneath the basemat was responded and described in the FSAR.

Question 03.08.05-22

Based on the NRC staff review, the applicant is requested to respond to the following:

a) The applicant revised Sections 3.8.5.5.4 and 3.8.5.6.3 to describe *Average Bearing Pressure* and *Localized Bearing Pressure* for the RXB and CRB. The applicant provided *Average Bearing Pressure* values of 4.6 ksf and 2.3 ksf for RXB and CRB, respective. However, the applicant did not provide any values for the *Localized Bearing Pressures* for RXB and CRB. Furthermore, in Item c, the applicant referred to Section 3.8.5.6.7 *Base Soil Pressure along Basemat Edges (Toe Pressure)* that referred to Tables 3.8.5-14 and 3.8.5-16 for the average toe pressures for the RXB and CRB. It is not clear to the staff whether *Localized Bearing Pressure* in Section 3.8.5.5.4 and *Basemat Soil Pressure along Basemat Edges (Toe Pressure)* in Section 3.8.5.6.7 are the same. Therefore, the applicant is requested to describe the difference, if any, between *Localized Bearing Pressure* and *Basemat Soil Pressure along Basemat Edges (Toe Pressure).*

b) Acceptable.

c) The NRC staff identified the following:

    i) In its response, the applicant did not provide any information related to the CRB Tunnel. Even though, the new Figure 3.8.5-3a, "Dynamic Pressure Contours on CRB basemat (psi)," seems to depict the worst pressure levels at the CRB Tunnel basemat. Therefore, the applicant is requested to describe analysis, design and associated tables and figures related to the CRB tunnel basemat.

    ii) The applicant did not provide any figures providing static base pressure contour, moment contours, etc. for the CRB basemat and CRB tunnel basemat. Therefore, the applicant is requested to provide figures providing static base pressure contour, moment contours, etc. for the CRB basemat and CRB tunnel basemat.

iii) EDITORIAL - for consistency, the applicant is requested to include the units into the titles of figures, as shown below, e.g.:

- Figure 3.85-2 RXB (psi)
- Figure 3.85-3 RXB (psi)
- Figure 3.85-4 M22 RXB (Kips-ft.)
- Figure 3.85-5 M11 RXB (Kips-ft.)
- Figure 3.85-6 M22 RXB (Kips-ft.)
- Figure 3.85-7 M11 RXB (Kips-ft.)

Staff feedback on Request for Additional Information 8901

Question 03.09.05-2

In the response provided, the applicant indicated that the steam generator (SG) flow restrictors are not pressure boundary components that are not classified as part of reactor internals. Furthermore, the applicant indicated that there is no design code associated with the SG flow restrictors. Design Control Document (DCD) Tier 2, Chapter 5, Figure 5.4-8 shows that the SG flow restrictors are located inside of the feed plenum tubesheets, which forms part of the reactor coolant pressure boundary (RCPB) and is designed to American Society of Mechanical Engineers (ASME) Subsection NB. It is unclear to the staff why the SG flow restrictors are not at least designed to ASME Code Subsection NG as a guide. By virtue of its location, the structural integrity of the SG flow restrictors can directly impact the structural integrity of the feed plenum tubesheet, and by extension, the SG tubes. Therefore, the applicant is requested to provide justification for not using any ASME design code for the SG flow restrictors, and to provide explanation on how the failure of the SG flow restrictors will impact the feed plenum tubesheet and SG tubes.

Question 03.09.05-3

In the response, the applicant provided explanation to the control rod drive (CRD) shaft supports. There are a total of five supports that are welded to the inside of the upper riser shell. These supports provide support to both the CRD shafts and the in-core instrumentation (ICI) guide tubes. In is unclear to the staff what is the tolerance between the CRD shafts and ICI guide tubes to the holes at which they are located in the supports. The applicant is requested to provide any wear mechanism at which these CRD shafts and ICI guide tubes are subjected to, as well as any inspection program, similar to the inspection program in MRP-227 for guide tube guide card wear, to ensure the CRD shafts will continue to function properly over time.

Question 03.09.05-4

In the response provided, the applicant stated that in the cold condition, the bellows applies approximately 500 pounds (lbs.) to the lower riser interface. Final design of the bellows in not yet completed. The applicant is requested to describe how this 500 lbs. will vary under different operating conditions, especially during high seismic events.

In the response, the applicant also stated that the annulus between the upper riser and vessel wall contains the steam generator tubes and tube supports. The upper riser is supported radially by the SG tube supports. The applicant is requested to describe how the upper riser is supported radially by the steam generator tube supports and to provide any stress and deflection limit for the upper riser and steam generator tube supports under both normal and accident conditions. In addition, the applicant did not address how the upper riser can behave like a vertical cantilever attached at the top from its original response.

The applicant's response also stated that a CRD shaft alignment drop test is to be conducted to determine the displacement limits for the CRD shaft supports. DCD Tier 2, Section 1.5.1.12 stated that the testing program is currently underway. The applicant is requested to provide the test result of this testing program, if the test result is not yet available, the applicant is requested to provide a schedule on when this test will be completed.

Question 03.09.05-5

In the response, the applicant stated that the control rod assembly (CRA) guide tube support plate is a grid structure with circular openings for the guide tubes. The applicant is requested to provide further detailed information on how the guide tubes are attached/secured to the CRA guide tube support plate. In addition, the applicant is requested to clarify if the guide tubes are removable. In operating pressurized-water reactor fleet, guide tubes are occasionally removed and interchanged at different locations when it’s necessary to even out guide tube guide card wear. The applicant is requested to explain if such maneuver is expected for the NuScale Power, LLC guide tubes.

In addition, the applicant is further requested to provide additional detailed information on the lock plate assemblies and the detailed design of the ball detents. The four slots shown in DCD Tier 2, Figure 3.9-3 and TR-0716-50439-P, Rev. 0, Figure 2-16 appear to be for alignment with the core support assembly. The applicant is requested to explain how the upper core plate is attached to the shell of the lower riser assembly (LRA).

Question 03.09.05-6

In the response, the applicant provided description of the CRA guide tubes. Specifically, the applicant explained that the top of the CRA guide tube assembly fits into a counter bore, with a slip fit, in the lower side of the CRA guide tube support plate. The applicant is requested to provide further clarification on how the top of the guide tube assembly is secured to the lower side of the CRA guide tube support plate by a slip fit. In addition, the applicant is requested to explain how thermal growth can affect this slip fit under different operating conditions and the relative movement between the guide tube assembly and the CRA guide tube support plate. It is also unclear to the staff how the CRA lower flange fit over the fuel pin caps and what mechanism is in place to prevent any relative movement between the CRA assembly and the lower core plate.

Question 03.09.05-7

In the response, the applicant provided detailed explanation of the ICI guide tubes. Several slip fit connections were used at different locations for the ICI guide tubes. The applicant is requested to provide information on how thermal expansion under different operating conditions will affect the clearance of all slip fit connections.

Question 03.09.05-8

In the response, the applicant explained that there are four upper support blocks welded to the core barrel spaced at 90 degree intervals. The upper support blocks are classified as core support structures because they transfer horizontal loads to the pressure vessel wall. The blocks fill the space between the core barrel and the reactor pressure vessel. The applicant is requested to provide information of the clearance gap between the upper support blocks and the insider wall of the reactor pressure vessel and how this clearance gap is affected by thermal expansion under all operating and accident conditions. In addition, the applicant is requested to clarify if all four upper support blocks are identical. Also, clarify the weld, whether it's ASME Subsection NB, of the core barrel guide to the reactor pressure vessel wall. Explain in detail the locking mechanism to couple the core support assembly to the lower riser as stated in paragraph 2 of the initial response.

The NRC staff understands that the core support assembly sits on the core support blocks, but it is unclear to the staff how the core support assembly is secured to the upper support blocks. Therefore, the applicant is requested to explain the mechanism in which the core support assembly is secured to the upper support blocks and how core support assembly lift off is prevented under severe seismic events.

Regarding the reflective block, since they are not fastened or physically connected to the core barrel as stated in the response, the applicant is requested to describe what mechanism is put in place to prevent excessive movement of the reflector blocks that may impact the fuel assemblies.

Question 03.09.05-10

In the response, the applicant explained that the core bypass flow is through two paths, the cooling channels in the reflector blocks, and the fuel assembly guide tubes and instrument tubes. DCD Tier 2, Section 4.4.3.1.1 indicates an additional path for core bypass flow, gap between the reflector blocks and core barrel. The applicant is requested to clarify the difference between the request for additional information (RAI) response and DCD Tier 2 Section 4.4.3.1.1 regarding core bypass flow.

Question 03.09.05-12

In the response, the applicant provided information on the load paths from the core barrel to the RPV. The applicant also stated that Beleville washer stacks are used in conjunction with the core support blocks. The NRC staff also reviewed RAI 8911 and the corresponding markups to TR-0916-51502. The response, the applicant stated that the lateral loads between the lower core plate and the core support blocks are transmitted in the circumferential direction through the stud that is integral with each core support block, which extends through the slotted holes in the lower core plate. However, the applicant also stated that the radial load path from the core barrel is through the upper support blocks attached at the top end of the core barrel. The applicant is requested to provide clarification that in the radial and circumferential directions, which load path is expected to have the highest load. In addition, Figure C-9 of the TR-0916-51502 markup in RAI 8911 only provided a simple sketch of the stud that extends through the slotted holes in the lower core plate, but it is unclear to the staff what is the detailed design of this stud, it's clearance between the Belleville washer stacks, and how it is loaded during a seismic event with lateral core barrel movement. The applicant is requested to provide more detailed information as stated above.

Question 03.09.05-13

In the response, the applicant stated that during refueling, the LRA is located in one of two configurations. To gain access to the fuel, the LRA may be attached from the core support assembly and lifted using the LRA lugs and stored in a designated stand in the refueling pool while the LRA is in the stand, it is supported by load bearing features that prevent the loading of the fuel pins or in-core instrumentation guide tubes that protrude below the upper core plate. The LRA stand is not a safety-related component, its detailed design has not yet been completed.

The applicant is requested to provide detailed information of the LRA lifting lugs, and what mechanism is put in place to prevent damage to the fuel pins when the LRA is in the LRA stand.

Question 03.09.05-15

In the response, the applicant provided information regarding deflection limits. Specifically, deflection limits are imposed on the components such as CRA guide tubes to assure the CRD shaft is sufficiently aligned so that the capability to insert the CRAs is not compromised. CRA drop and CRD shaft alignment testing will provide data to support determination of specific values for the maximum deflections that allow CRA insertion requirements to be met. The applicant is requested to provide the CRA drop and CRD shaft alignment testing result; if such result is not currently available, include its scheduled completion date. In addition, the applicant is requested to provide the detailed design criteria used to determine the deflection limits once the CRA drop and CRA shaft alignment testing are completed.