NuScaleTRRaisPEm Resource

From:	Tabatabai-Yazdi, Omid
Sent:	Friday, October 07, 2016 10:57 AM
To:	'Tom Bergman'
Cc:	Mirsky, Steven; 'Steven Pope'; 'Steve Unikewicz'; NuScaleTRRaisPEm Resource; Tonacci, Mark; 'Wike, Jennie'
Subject:	NRC RAI Letter No. 8 for the review of Topical Report, "Safety Classification of Passive Nuclear Power Plant Electrical Systems"
Attachments:	NuScale RAI Letter No. 8 - Electrical Topical Report.pdf

Dear Mr. Bergman,

Attached find please NRC staff's request for additional information for subject topical report. Please submit your response to the NRC Document Control Desk by December 7, 2016. If you have any questions, please feel free to contact me.

Kind regards,

Omid Tabatabai, Senior Project Manager
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Subject: NRC RAI Letter No. 8 for the review of Topical Report, "Safety Classification of Passive Nuclear Power Plant Electrical Systems"
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From: Tabatabai-Yazdi, Omid

Created By: Omid.Tabatabai@nrc.gov

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Expiration Date:
Recipients Received:

October 7, 2016

Mr. Thomas Bergman
Vice President, Regulatory Affairs
NuScale Power, LLC
1100 NE Circle Boulevard, Suite 200
Corvallis, OR 97330

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 8 FOR THE REVIEW OF TOPICAL REPORT 0815-16497, "SAFETY CLASSIFICATION OF PASSIVE NUCLEAR POWER PLANT ELECTRICAL SYSTEMS," REVISION 0. (CAC NO. RQ6002)

Dear Mr. Bergman:

In an October 29, 2015, letter NuScale Power, LLC, (NuScale) submitted Topical Report (TR) 0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems," Revision 0 (Agencywide Documents Access and Management System (ADAMS) Accession ML15306A126) for the U.S. Nuclear Regulatory Commission (NRC) staff's review and approval. The NRC staff is performing a detailed review of this topical report and has identified that additional information is needed to continue portions of the review. The NRC staff's request for additional information (RAI) is contained in the enclosure to this letter.

On July 22, 2016, the NRC staff provided a draft version of the aforementioned RAI questions to NuScale and subsequently on August 8, 2016, held a teleconference with NuScale to ensure that the RAI questions were clear prior to formally submitting them to NuScale. During this call, NuScale pointed out that the RAIs were not consistent with the intent of the topical report because they were written based on the NuScale electrical system design. Specifically, NuScale requested that the staff write the RAI questions not only for NuScale small modular reactor but also for any passive plant, including advanced reactor design. As a result of the August 8th teleconference, the NRC staff modified the draft RAI questions to add clarity to some questions based on the feedback received from NuScale. On September 1, 2016, the staff emailed you a copy of the revised RAI questions. On September 7, 2016, you requested that the NRC not proceed with formally issuing the RAIs until NuScale has had a chance to provide additional feedback on the revised RAI questions. On September 20, 2016, Mr. Steven Mirsky, NuScale, provided additional comments on the staff's revised RAI questions before a second clarification teleconference with the staff on October 6, 2016 (ADAMS Accession ML16281A103.) On October 3, 2016, the NRC technical staff involved in the review of your topical report, as well as the project management team, internally met to review and discuss NuScale's additional comments on revised RAIs. As a result of the internal meeting, the staff concluded that it did not agree with the proposed NuScale comments and that the staff RAI questions would be issued as drafted. If additional clarification is needed, the staff is available to discuss your questions in a public meeting.

To support the review schedule, NuScale is requested to respond within 60 calendar days of the date of this letter. If changes are needed to the topical report, the NRC staff requests that the RAI response include the proposed wording changes.

If you have any questions or comments concerning this matter, you may contact me at 301-415-6616.

Sincerely,

/RA/

Omid Tabatabai, Senior Project Manager
Licensing Branch 1
Division of New Reactor Licensing
Office of New Reactors

Docket No. PROJ0769
eRAI Tracking Nos. 8607, 8632, 8669

Enclosure: Request for Additional Information Letter No. 8

Request for Additional Information Letter No. 8

Topical Report 0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems”
Review Section: 08.03.02

The electrical power system presented in the Topical Report (TR) depicts a design with no Class 1E power sources as the proposed reactor design does not require any safety-related electrical loads to support the safety analyses. However, 10 CFR 50.34(f)(2)(xx) calls for vital-bus-powered post- accident monitoring instrumentation with backup power from emergency power supplies. In order for the staff to be able to conclude that an electrical design such as the one presented in the TR provides equivalent protection to that prescribed in the regulation, the staff must be able to conclude that the proposed design is of similar (high) reliability. To that end, the staff requires the following additional information:

08.03.02-01

Table 3-2 of the TR states that Valve Regulated Lead Acid (VRLA) batteries will be used for the direct current (DC) power system. Based on various industry publications, including IEEE Std. 1187, “Recommended Practice for Installation Design and Installation of Valve-Regulated Lead-Acid (VRLA) Batteries for Stationary Applications,” the life of a VRLA battery can be seriously and suddenly reduced due to factors such as: 1) prolonged high ambient temperatures, 2) magnitude and frequency of discharge cycles, and 3) overcharging. Please describe how these factors will be addressed in the design and operation of a passive reactor nuclear power plant that relies on VLRA battery systems to ensure high reliability DC power system.

08.03.02-02

Table 3-2 of the TR provides a comparison of the “Class 1E DC Electrical system” to the “Non Safety-Related DC Electrical System(s) Relied upon to Power Type B and Type C Accident Monitoring Instrumentation.” Under the provision “Quality Assurance” in the Table 3-2, it stated that a Graded QA Program will be applied to the DC Electrical System, which will meet or exceed the augmented QA provisions specified in RG 1.155, Appendix A, “Quality Assurance Guidance for Non-Safety Systems and Equipment”. RG 1.155, Appendix A provides QA guidance for meeting the requirements of 10 CFR 50.63 and not already explicitly covered by existing QA requirements in 10 CFR Part 50 in Appendix B or R. Please describe the proposed quality assurance program in sufficient detail that will allow the staff to verify it meets or exceeds the provisions of RG 1.155.

08.03.02-03

Table 3-2 of the TR, under the provision “Batteries,” states that the VRLA batteries have augmented design, QA, and qualification provisions. Please describe the methods and processes that will be used by a passive reactor nuclear power plant to verify that VRLA batteries will perform their intended function(s) during normal operation, operational occurrences and

postulated design basis events. Please also provide the industry standards or applicable references that will be used for verification purposes.

08.03.02-04

The TR describes the presented dc power system as “highly reliable” and substantially equal in reliability to that of an analogous Class 1E dc power system. These statements have not been described adequately in the TR. In order for the staff to be able to fully evaluate the design and ultimately conclude on its acceptability as a highly reliable power system, the staff requests that NuScale provide a description of the methodology that will be used to compare the highly reliable DC system to be described in its design certification application to a Class 1E dc power system to show that the highly reliable DC system is substantially equal in reliability to a typical Class 1E dc power system.

08.03.02-05

The regulation set forth in 10 CFR 50.55a(h)(3) requires that design certification applications under part 52 meet the requirements of IEEE Std. 603-1991, “Criteria for Safety Systems for Nuclear Power Generating Stations.” IEEE Std. 603-1991 provides a definition of "safety system" and states that the electrical portion of the safety systems, that perform safety functions, is classified as Class 1E. Included in the definition of safety system is a system that is relied upon to remain functional during and following a design basis event to ensure the capability to shut down the reactor and maintain it in a safe shutdown condition.

Condition of Applicability Item I.1.b, contained in Table 3-1 of the TR, states that sufficient reactor coolant inventory and negative reactivity are assured during and following a design basis event to achieve and maintain safe shutdown. Additionally, the TR provides a clarifying example assessment to illustrate how the Conditions of Applicability would be demonstrated. This example assessment did not include a quantitative safety analysis to demonstrate the ability to insert sufficient negative reactivity during and following a design basis event to achieve and maintain safe shutdown.

SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” clarifies the conditions that constitute a safe shutdown as reactor subcriticality, decay heat removal, and radioactive material containment. Additionally, SECY 94-084 states that an appropriate safety analysis can be used to demonstrate passive system capabilities to bring the plant to a safe stable condition and to maintain this condition. NRC staff is seeking to clarify whether Condition of Applicability Item I.1.b is consistent with the description of safe shutdown provided in SECY-94-084. Additionally, NRC staff is seeking to clarify the requirements for demonstrating how Condition of Applicability Item I.1.b is satisfied. NRC staff requests the following additional information:

1. Specify the criteria that constitute a safe-shutdown as applied to Condition of Applicability Item I.1.b.

2. Describe how a future passive plant applicant will demonstrate that electrical power is not necessary to achieve and maintain a safe shutdown for a minimum of 72 hours.

08.03.02-06

GDC 15 requires the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Condition No. I.1 of the Conditions of Applicability, contained in Table 3-1 of TR-0815-16497, states that for a design basis event, electrical power is not necessary to maintain the reactor coolant pressure boundary (RCPB) integrity for a minimum of 72 hours. Additionally, TR-0815-16497 provides a clarifying example assessment to illustrate how the Conditions of Applicability would be demonstrated. This example assessment includes a safety analysis showing an example passive plant response to an anticipated operational occurrence. The safety analysis shows that the example passive plant response to the anticipated operational occurrence includes establishing a direct coolant flow path between the reactor core and the containment, thereby removing a fission product barrier. This caused NRC staff to question if the items under Conditions of Applicability I.1 are sufficient to demonstrate RCPB integrity. Additionally, RIS 2005-29, discusses the design criteria for event non-escalation. NRC staff is questioning why the removal of a fission product barrier is not considered an event escalation.

NRC staff requests the following information:

1. Specify the criteria that constitute RCPB integrity as applied to Condition No. I.1 of the Conditions of Applicability.

2. Explain why the removal of a fission product barrier during an anticipated operational occurrence is not considered an event escalation.