NuScaleDCRaisPEm Resource

From: Chowdhury, Prosanta
Sent: Thursday, May 10, 2018 9:38 AM
To: Request for Additional Information
Cc: Lee, Samuel; Cranston, Gregory; Franovich, Rani; Karas, Rebecca; Thomas, Matt; NuScaleDCRaisPEm Resource
Subject: Request for Additional Information No. 468 eRAI No. 9420 (15)
Attachments: Request for Additional Information No. 468 (eRAI No. 9420).pdf

Attached please find NRC staff's request for additional information (RAI) concerning review of the NuScale Design Certification Application.

The NRC Staff recognizes that NuScale has preliminarily identified that the response to one or more questions in this RAI is likely to require greater than 60 days. NuScale is expected to provide a schedule for the RAI response by email within 14 days.

If you have any questions, please contact me.

Thank you.

Prosanta Chowdhury, Project Manager
Licensing Branch 1 (NuScale)
Division of New Reactor Licensing
Office of New Reactors
U.S. Nuclear Regulatory Commission
301-415-1647

Hearing Identifier:	NuScale_SMR_DC_RAI_Public
Email Number:	499

Mail Envelope Properties (DM6PR09MB261841F2B8B0EC42D0F00C5D9E980)

Subject:	Request for Additional Information No. 468 eRAI No. 9420 (15)
Sent Date:	5/10/2018 9:38:26 AM
Received Date:	5/10/2018 9:38:30 AM
From:	Chowdhury, Prosanta

Created By: Prosanta.Chowdhury@nrc.gov

Recipients:
"Lee, Samuel" <Samuel.Lee@nrc.gov>
Tracking Status: None
"Cranston, Gregory" <Gregory.Cranston@nrc.gov>
Tracking Status: None
"Franovich, Rani" <Rani.Franovich@nrc.gov>
Tracking Status: None
"Karas, Rebecca" <Rebecca.Karas@nrc.gov>
Tracking Status: None
"Thomas, Matt" <Matt.Thomas@nrc.gov>
Tracking Status: None
"NuScaleDCRaisPEm Resource" <NuScaleDCRaisPEm.Resource@nrc.gov>
Tracking Status: None
"Request for Additional Information" <RAI@nuscalepower.com>
Tracking Status: None

Post Office: DM6PR09MB2618.namprd09.prod.outlook.com

Files	**Size**	**Date & Time**
MESSAGE	675	5/10/2018 9:38:30 AM
Request for Additional Information No. 468 (eRAI No. 9420).pdf		28440

Options

Priority:	Standard
Return Notification:	No
Reply Requested:	No
Sensitivity:	Normal
Expiration Date:	
Recipients Received:	

Request for Additional Information No. 468 (eRAI No. 9420)

Issue Date: 05/10/2018
Application Title: NuScale Standard Design Certification - 52-048
Operating Company: NuScale Power, LLC
Docket No. 52-048
Review Section: 15 - Introduction - Transient and Accident Analyses
Application Section:

QUESTIONS

15-17

Regulatory Basis

General Design Criterion (GDC) 1 in 10 CFR Part 50, Appendix A, requires structures, systems, and components (SSCs) important to safety to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. The NRC regulations in 10 CFR 50.2 define safety-related, in part, as SSCs that prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of that chapter, as applicable. In addition, 10 CFR 52.47(a)(2)(iv) provides equivalent siting and safety analysis offsite dose guidelines for new nuclear power plant standard design certifications.

Introduction

In RAI 8744, Question 15.02.08-4, RAI 9205, Question 15-3, and RAI 9237, Question 15.06.03-3 the NRC staff requested the applicant to provide additional information justifying the credit of nonsafety-related components for design basis accident (DBA) mitigation. The applicant's response referred to the guidance in RG 1.206 which specifies that nonsafety-related components may be used as backup protection to mitigate transient or accident conditions; or NUREG-0138 which details the acceptable means by which nonsafety-related components can be credited for DBA mitigation. The staff found the applicant's response generally acceptable; however, the staff also determined that more information is required in order to find that the applicant is appropriately meeting the guidance provided in NUREG-0138, precedence, and overall find that the applicant's FSAR Tier 2, Chapter 15 analyses are in compliance with the applicable regulations.

Discussion

The NRC staff agrees that NUREG-0138 (November 1976), "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976, Memorandum from Director, NRR to NRR Staff," in Issue No. 1, "Treatment of Non-Safety Grade Equipment in Evaluations of Postulated Steam Line Break Accidents," discusses the acceptance of reliance on specific nonsafety-related valves as part of the mitigation of secondary line breaks.

In particular, NUREG-0138 defines the issue as follows:

In evaluating the consequences of postulated breaks of steam lines the current staff position (SRP 10.3) states that the design should preclude the blowdown of more than one steam generator, assuming a concurrent single component failure, and assuming that the turbine stop and control valves remain functional. Provided that these valves and their control systems are designed for closure under the postulated conditions, and because they are high quality components, the staff does not require that they be designed to the requirements for safety-related equipment.

Regarding the reliance on non-safety grade equipment[1] to mitigate steam line break accidents, NUREG-0138 includes the following general NRC staff position:

For loss-of-coolant accidents (LOCA) involving a spontaneous rupture of the primary system boundary, where significant damage to the fuel and a major release of fission products are potential consequences, the most stringent quality and design requirements, including seismic qualification, are imposed on those systems needed to prevent and cope with a LOCA. However, for accidents involving spontaneous failures of secondary system piping not part of the primary system boundary, where the potential consequences are significantly lower, less stringent requirements are imposed on the quality and design of the systems needed to cope with such secondary system ruptures. This approach results, in the staff's judgment, in a proper weighing of consequences and safety requirements in order to assure a balanced level of safety over the entire spectrum of postulated design basis accidents.

In NUREG-0138, the NRC staff discusses the reliance on non-safety grade valves, such as turbine stop, control, and intercept valves to mitigate the consequences of a steam line break. For example, the staff indicates in NUREG-0138 that the continued reliability of these components over the life of the plant is assured by frequent (generally weekly) inservice tests. The staff also states that the NRC conducted a survey of the reliability of these valves at operating light water reactors. The staff found no control system failures and few incidents where the valves did not fully close. Based on its review, the staff concluded that the reliability of these valves is of the same order of magnitude as that accepted for nuclear safety-grade components. Regarding feedwater isolation, the staff states that the rationale for reliance on "non-safety grade" feedwater components is similar to that presented for steam line valves. In NUREG-0138, the staff states its belief that it is acceptable to rely on these non-safety grade components in the steam and feedwater systems because their design and performance are compatible with the accident conditions for which they are called upon to function.

NUREG-0138 is referenced in other NRC and industry documents discussing reliance on nonsafety-related components as part of the mitigation of accidents. For example, in NUREG-1793 (September 2004), "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," the NRC staff states in Chapter 15, "Transient and Accident Analyses," Section 15.1.2, "Non-Safety-Related Systems Assumed in the Analysis," that crediting nonsafety-related backup systems and components in the design-basis analyses is acceptable for several reasons, including operating data that show that the turbine stop and control valves are reliable, and taking credit for the turbine valves in the design-basis analyses for backup protection is consistent with the staff position stated in NUREG-0138. In a request for additional information dated November 16, 2009 (ADAMS Accession No. ML093140231) regarding a proposed change to Technical Specifications for the auxiliary feedwater system at the Point Beach nuclear power plant, the NRC staff notes that NUREG-0138 allows a licensee to take credit for nonsafety-grade components in the main feedwater line, even though they are not designed seismic Category I, to perform a backup isolation function in certain accident scenarios, because the staff does not require that an earthquake be assumed to occur coincident with the postulated main steamline break. In its November 16, 2009, letter, the NRC staff further indicates that NUREG-0138 prescribes that in order to rely on these nonsafety-grade components, their design and performance must be compatible with the accident conditions for which they are called upon to be credited, and the reliability of these valves is of the same order of magnitude as that accepted for nuclear safety-grade components. In its reply dated December 16, 2009 (ADAMS Accession No. ML093510809), the Point Beach licensee provides information supporting its determination that the design and performance of the main feedwater regulating valves, which would be downgraded to nonsafety-related, will be compatible with the accident conditions for which the valves will be credited. These NRC and licensee documents indicate that the application of NUREG-0138 requires that the nonsafety-related components to be credited for main steam line or feedwater line breaks must be determined to be reliably capable of performing their intended function.

In its response to RAI 8744, Question 15.02.08-4, the NuScale design certification applicant states that the nonsafety-related FW supply check valves to be credited for a line break are listed in FSAR Tier 2, Section 3.2, "Classification of Structures, Systems, and Components," Table 3.2-1 of the same title, with requirements for augmented quality, designed to Seismic Category I and included in the inservice testing program. The NRC staff notes that Table 3.2-1 lists Feedwater Supply Check Valve (without an identifying valve number) as Seismic Category I with "Technical Specification Surveillance for operability and in-service testing" in the "Augmented Design Requirements" column. The "QA Program Applicability" column specifies AQ-S with Note 2 indicating that AQ-S "indicates that the pertinent requirements of 10 CFR 50 Appendix B are applicable to SSC classified as seismic category II in accordance with the quality assurance program."

In its response to RAI 9205, Question 15-3, the applicant provides information on the three nonsafety-related components that are credited in the NuScale Power Module Chapter 15 safety analyses (nonsafety-related feedwater check valves, nonsafety-related feedwater regulating valves, and nonsafety-related secondary (backup) main steam isolation valves (MSIV) and bypass valves). The information provided details the applicant's position on why the crediting of each valve is appropriate for DBA mitigation. As mentioned above, the staff generally finds the applicant's arguments acceptable; however, additional information is required to assure the applicant is appropriately meeting the guidance in NUREG-0138.

In its response to RAI 9237, Question 15.06.03-3, the applicant provided information related to the crediting of the nonsafety-related secondary MSIVs for a steam generator tube failure (SGTF). The response explains that due to the augmented design requirements identified, precedence from prior design certifications, and statements in RG 1.206, it is appropriate to credit the secondary MSIVs to mitigate the effects of an SGTF. However, the staff needs additional information in order to reach a safety finding on the applicant's proposal.

<u>Request</u>

For reliance on nonsafety-related components to mitigate a DBA in the NuScale accident analysis, consistent with the guidance in NUREG-0138, the NRC staff requests that the NuScale design certification applicant provide the following information:

(1) Identifying numbers for the applicable nonsafety-related components,

(2) Type of components (e.g., if a check valve, then what type - swing check or nozzle check, etc.),

(3) Performance history and operating experience for the applicable components and their application in the NuScale design,

(4) Detailed design and qualification requirements to be applied to the components, including the associated electrical design and qualification requirements where it is applicable (i.e., backup isolation valves that require electrical signals to actuate),

(5) Preservice and inservice testing, and Technical Specification requirements to be applied to the components,

(6) Planned modifications to the NuScale Design Certification application, such as Part 2 (FSAR Tier 1 and Tier 2), and Part 4 (Technical Specifications), to specify the design, qualification, ITAAC, preservice and inservice testing, and Technical Specification requirements for the applicable components, and

(7) Clarification of the reference to AQ-S in the QA Program Applicability column in NuScale FSAR Tier 2, Table 3.2-1, for the applicable nonsafety-related valves to be credited.

In addition, for reliance on the nonsafety-related secondary MSIV to mitigate an SGTF event, as required by the definition of safety-related in 10 CFR 50.2, the applicant must ensure that the estimated offsite doses do not exceed the requirements of 10 CFR 52.47(a)(2)(iv) if the secondary MSIV fails to close. Therefore, the applicant is requested to:

(8) Perform an analysis that demonstrates 10 CFR 52.47(a)(2)(iv) criteria are met assuming the secondary MSIV fails to close for the SGTF event.

With regards to the feedwater regulating valve (FRV), which is credited to close within a specified time in the containment response analysis, the imposed closure time in the analysis needs to be clearly reflected in the FSAR. Although Technical Specification Surveillance Requirement 3.7.2.1 indicates a surveillance to verify the closure time is within limits, that value should be specified in the FSAR. Accordingly, in addition to the aforementioned augmented quality considerations, the staff requests that NuScale:

(9) Provide a closure time for the FRV corresponding to that assumed in the analysis in the FSAR.

---

[1] In the past, mechanical equipment classified as safety-related was referred to as meeting safety-grade qualification requirements, such as seismic, environmental, and functional requirements.