NuScaleDCRaisPEm Resource

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Sent:	Wednesday, June 21, 2017 8:45 AM
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Subject:	RE: Request for Additional Information No. 69, RAI 8809
Attachments:	Request for Additional Information No. 69 (eRAI No. 8809).pdf

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

Please submit your response within 60 days of the date of this RAI to the NRC Document Control Desk.

If you have any questions, please contact me.

Thank you.

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301-415-0546

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Request for Additional Information No. 69 (eRAI No. 8809)

Issue Date: 06/21/2017
Application Title: NuScale Standard Design Certification - 52-048
Operating Company: NuScale Power, LLC
Docket No. 52-048
Review Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: 19

QUESTIONS

19-9

The scope, criteria, and process used to determine Regulatory Treatment of Nonsafety Systems (RTNSS) for the passive plant designs are established in:

1. SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Nonsafety Systems in Passive Plant Designs," dated March 28, 1994 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML003708068) and associated Staff Requirements Memorandum (SRM), June 30, 1994 (ADAMS Accession No. ML003708098);

2. SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Nonsafety Systems (RTNSS) in Passive Plant Designs," dated May 22, 1995 (ADAMS Accession No. ML003708005), and associated SRM, June 28, 1995 (ADAMS Accession No. ML003708019); and

3. SECY-96-128, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," June 12, 1996 (ADAMS Accession No. ML003708224), and associated SRM, January 15, 1997 (ADAMS Accession No. ML003755486).

The NRC staff uses guidance contained in Standard Review Plan (SRP) 19.3 Revision 0, dated June 2014, "Regulatory Treatment of Non-Safety Systems for Passive Advanced Light Water Reactors," to conduct its review of an applicant's RTNSS evaluation. In accordance with SRP 19.3, Revision 0 (page 19.3-6), staff responsible for the review of the applicant's probabilistic risk assessment (PRA) will verify that the applicant has met the following acceptance criterion: The applicant has determined those non-safety related structures, systems and components (SSCs), if any, used to prevent the occurrence of initiating events and, based on their importance to risk as determined from the PRA, has included them in the scope of RTNSS.

The staff has reviewed the information in Section 19.3 of the final safety analysis report (FSAR) and examined additional clarifying information from an audit of information supporting Chapter 19 of the FSAR and determined that it needs additional information to complete its review of Section 19.3 of the FSAR, as follows:

1. The staff could not verify that the applicant completely addressed the following screening criteria for assessing the risk of significance of Structures, Systems, and Components (SSCs) with respect to initiating event frequency as stipulated on page 19.3-10 of SRP 19.3, Revision 0, dated 2014:

a: Does the calculation of the initiating event frequency consider the nonsafety-related SSCs?
b. Does the unavailability of the non safety-related SSCs significantly affect the calculation of the initiating event frequency?
c. Does the initiating event significantly affect the CDF and the LRF -contributes more than 10 percent of the at power or shutdown internal events CDF?

The staff requests that the results of the above assessment be documented in the FSAR so that staff can make a reasonable assurance finding.

B. The staff could not find an evaluation for determining non-safety related SSCs that could prevent a module drop Initiating Event during refueling operations. Such SSCs may include, but not be limited to (1) DC power and (2) the Heavy Load System, which includes the reactor building crane (RBC), the RBC control system devices, Containment Vessel Flange Tool, the Reactor Vessel Flange Tool, and the Module Lift device.   The staff is requesting the applicant to provide this evaluation in the FSAR or justify in the FSAR why such an evaluation is unnecessary.

19-10

10 CFR 52.47(a)(2) states that it is expected that the standard plant will reflect through its design, construction, and operation an extremely low probability for accidents that could result in the release of radioactive fission products. 10 CFR 52.47(a)(4) states that each design certification application must contain an FSAR that includes an analysis and evaluation of the design and performance of SSCs with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents. 10 CFR 52.47(a)(27) states that a DC application must contain an FSAR that includes description of the design-specific PRA and its results. SRP Chapter 19.0, Revision 3, states, "The Commission approved the staff’s position that advanced [light water reactor] vendors should perform bounding analyses of site-specific external events likely to be a challenge to the plant (such as river flooding, storm surge, tsunami, volcanism, high winds, and hurricanes). When a site is chosen, its characteristics should be compared to those assumed in the bounding analyses to ensure that the site is enveloped."

Regarding the Tornado and Hurricane Analysis, specified in the FSAR, in Plant Operating State (POS) 3, POS 4, POS 5, and POS 6, it is assumed that operators will not transport a module under a hurricane warning or conditions when a tornado strike is likely. This assumption is the basis for tornados and hurricanes not being evaluated during module movement in Chapter 19 of the FSAR. This assumption is also the basis for the staff reaching a reasonable assurance finding for the risk impact of high winds on module movement risk. The staff requests the applicant to justify in the FSAR how this PRA assumption will be maintained by the combined license holder (e.g., by a Limiting Condition of Operation, a Condition of the License, or some other means).

19-11

10 CFR 52.47(a)(2) states that it is expected that the standard plant will reflect through its design, construction, and operation an extremely low probability for accidents that could result in the release of radioactive fission products. 10 CFR 52.47(a)(4) states that each DC application must contain an FSAR that includes an analysis and evaluation of the design and performance of SSCs with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents. 10 CFR 52.47(a)(27) states that a DC application must contain an FSAR that includes description of the design-specific PRA and its results. SRP Chapter 19.0 Revision 3, states, "The Commission approved the staff's position that advanced LWR vendors should perform bounding analyses of site-specific external events likely to be a challenge to the plant (such as river flooding, storm surge, tsunami, volcanism, high winds, and hurricanes). When a site is chosen, its characteristics should be compared to those assumed in the bounding analyses to ensure that the site is enveloped."

I1. In this context, the staff could not find a technical justification in the ASME PRA Standard or DC\COL Interim Staff Guidance 028 to support the probable maximum flood frequency, cited in reference 19.1-2 as 2.0 E-3 (with an error factor of 10), to genrically bound any US location for any potential flooding mechanism such as local intense precipitation or storm surge or dam failure, given that no site is being referenced. DC\COL Interim Staff Guidance 028 states, " DC applications will not have regional or site-specific information on which to base their analysis. Instead, DC applicants are expected to establish site characteristics and site-interface requirements to generically bound or represent the analysis. At the COL application stage, site-specific information is available and can be used directly or in confirming the DC analysis'".

2. The staff could not find a technical basis for the following assumption from the FSAR, " For 90 percent of flood events, operators are assumed to cease refuelingand crane operations, and perform a controlled shutdown prior to potentialxternal flood-induced equipment impacts (e.g., due to LOOP), when forecasts or conditions indicate the potential for SSC susceptibility to an external flood. The remaining 10 percent of floods are assumed to result in a LOOP while the plant is still at power, when AC power is lost to plant transformers and power production loads such as the feedwater pumps and condensate pumps.". In absence of a specific site or even a specific flooding mechanism,this assumption is not generically bounding, but appears in the dominant cutsets. The staff requests that this assumption be removed from the external flooding PRA or that the applicant provide a technical justification for this assumption.

3. The staff could not find an evaluation or discussion regarding the potential failure of flooding penetrations or even which specific flooding penetrations (such as flood doors above and below grade) are risk significant. The staff requests Nuscale to identify risk significant flooding penetrations in Chapter 19 of the FSAR.

19-12

10CFR 52.47(a) states, "The application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information."

10 CFR 52.47(a)(11) states, "Proposed technical specifications prepared in accordance with the requirements of §§ 50.36 and 50.36a of this chapter."

10 CFR 50.36(c)(2)(ii), states, "A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:
(D) Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety."

Federal Register, Vol. 60, No. 138, Wednesday, July 19, 1995, Rules and Regulations, 36953, for the Technical Specifications Final Rule, states, "The Commission identified four systems that meet Criterion 4 based on previous qualitative reviews of operating experience and risk. They are reactor core isolation cooling/isolation condenser, residual heat removal, standby liquid control, and recirculation pump trip. The Commission recognizes however, that other structures, systems, or components may meet this criterion. Plant and design specific PRAs have yielded valuable insights to unique plant vulnerabilities not fully recognized in the safety design basis accident, or transient analyses."

The staff reviewed technical report, "Technical Specifications (TS) Regulatory Conformance and Development" TR-1116-52011. The staff reviewed the applicant's evaluation of potential Structures, Systems, and Components (SSCs) that meet Criterion 4 of 10CFR50.36 on the basis of risk and operating experience. Since no non-safety related SSC or functions approached the 10E-6 reactor year criterion, no non safety related SSCs were included in TS on the basis of risk. The risk evaluation seemed to focus on full power risk core damage frequency. There was no discussion on shutdown risk, module drop risk, and external events risk. The staff requests a summary of the risk evaluation in Chapter 19 of the FSAR on possible SSCs that meet Criterion 4 of 10CFR50.36 on the basis of risk that includes: full power risk, shutdown risk, module drop risk, external events risk, and multi-module considerations.