From: Chowdhury, Prosanta
Sent: Monday, April 26, 2021 10:18 AM
To: NuScaleTRRaisPEm Resource
Cc: Schiller, Alina
Subject: FW: Request for Additional Information No.9828 (eRAI No. 9828)
Attachments: Request for Additional Information 9828 (eRAI No. 9828).docx

From: Chowdhury, Prosanta
Sent: Thursday, April 22, 2021 4:46 PM
To: Request for Additional Information <RAI@nuscalepower.com>
Cc: Dudek, Michael <Michael.Dudek@nrc.gov>; Cranston, Greg <Gregory.Cranston@nrc.gov>; Schiller, Alina <Alina.Schiller@nrc.gov>; Pohida, Marie <Marie.Pohida@nrc.gov>; Neuhausen, Alissa <Alissa.Neuhausen@nrc.gov>; Rosenberg, Stacey <Stacey.Rosenberg@nrc.gov>; English, Liz <EEnglish@nuscalepower.com>; Luchsinger, Deb <DLuchsinger@nuscalepower.com>
Subject: Request for Additional Information No.9828 (eRAI No. 9828)

Attached please find NRC staff's request for additional information (RAI) concerning the review of Licensing Topical Report TR 0915-17772, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones," Revision 2 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20217L422).

Please submit your technically correct and complete response by May 19, 2021, to the NRC Document Control Desk.

If you have any questions, please do not hesitate to contact me or Alina Schiller (copied).

Thank you.

PROSANTA CHOWDHURY, PROJECT MANAGER
US NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
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From: Chowdhury, Prosanta

Created By: Prosanta.Chowdhury@nrc.gov

Recipients:
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Tracking Status: None
"NuScaleTRRaisPEm Resource" <NuScaleTRRaisPEm.Resource@nrc.gov>
Tracking Status: None

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Request for Additional Information 9828 (eRAI No. 9828)
Issue Date: 04/19/2021
Application Title: NuScale Topical Report
Operating Company: NuScale Power, LLC
Docket No. 99902043
Review Section: 01.05 - Other Regulatory Considerations

QUESTIONS

01.05-43

The following regulatory basis and discussion applies to all six questions in this request for additional information (RAI).

Regulatory Basis:

Title 10 of the *Code of Federal Regulations* (CFR) 50.47 and 10 CFR Part 50 Appendix E codify the emergency planning requirements for nuclear power reactors. Specifically, the plume exposure emergency planning zone (EPZ) for power reactors generally consists of an area about 10 miles in radius for reactors with an authorized power level greater than 250 megawatts thermal (MWt), and the EPZ may be determined on a case-by-case basis for reactors with an authorized power level less than 250 MWt. If a reactor has an authorized power level greater than 250 MWt, an exemption may be required for an EPZ less than 10 miles in radius. The technical basis for the 10-mile plume exposure EPZ is given in NUREG-0396/EPA 520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants" (Agencywide Document Access and Management System (ADAMS) Accession No. ML051390356), and was based upon evaluation of the offsite consequences of accidents (both design basis and severe) and a comparison of doses to the Environmental Protection Agency (EPA) guidance on when to take emergency response actions. The EPA emergency response actions include sheltering and evacuation as given in the Protective Action Guides (PAGs), or, for very low-probability and high-consequence accidents, demonstration that the probability of exceeding a radiation exposure deterministic health effect is low and decreasing at the chosen outer boundary of the plume exposure EPZ. The assumptions and approach used in the analysis of the NuScale EPZ sizing methodology, including the selection of accident sequences for source term calculations, can impact the results.

Discussion

NuScale Power, LLC (NuScale) submitted licensing topical report (LTR) TR-0915-17772-P, Revision 2, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones" for review by the NRC staff. As stated in Section 3.0, "Accident Screening Methodology," of the LTR, a combined license (COL) applicant can use this risk-informed approach to select appropriate single and multi-module accident sequences for the EPZ technical basis based on their Probabilistic Risk Assessment (PRA) which is expected to cover all internal and external initiators. The Executive Summary further states that, "The methodology is intended for use by all advanced nuclear reactor designs, particularly small modular reactors (SMRs) such as NuScale, although it is acknowledged that not every element of the method will be applicable to all designs."

Issue

The Commission Policy Statement, "Safety Goals for the Operations of Nuclear Power Plants" (Federal Register Notice 51 FR 28044) established Quantitative Health Objectives (QHOs), which broadly define an acceptable level of radiological risk to the public from nuclear power plant operation. The QHOs have been translated into two numerical objectives:

- the individual risk of prompt fatality from a reactor accident (includes the aggregate of possible reactor accidents) should be less than 5E-7 per reactor-year (ry). The "vicinity" of a nuclear power plant is understood to be a distance extending to 1 mile from the plant site boundary;

- the risk of cancer to the population in the area near a nuclear power plant due to its operation should be limited to 2E-6 per ry. The "area" is understood to be an annulus of 10-mile radius from the plant site boundary.

According to the LTR, Section 3.4.3, "Screening of Single Module Accident Sequences on Core Damage Frequency," the COL applicant would perform accident sequence screening based on frequency. First, all external event sequences with an initiating event frequency less than 1E-5 per year would screen out. Second, all internal and remaining external events with a core damage sequence frequency less than 1E-7 per year would screen out before performing consequence analysis.

Request

The staff requests that the applicant justify in the LTR how the methodology confirms that the aggregate of the screened-out sequences do not cause the QHOs to be exceeded or provide revised screening criteria which provides this justification.

01.05-44

Issue

Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (ADAMS Accession No. ML17317A256) states that "all plant operating modes and hazard groups be addressed when those risk contributions affect the decision." The ASME/ANS PRA standard addresses internal flood, internal fire, seismic, wind, external flood and other external hazards. The 1995 PRA Policy Statement states: "PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review." In the Staff Requirements Memorandum (SRM) to SECY-04-0118, "Plan for the Implementation of the Commission's Phased Approach to Probabilistic Risk Assessment Quality," states that "the licensee's submittal is expected to be in conformance with the published standards."

According to the LTR, Section 3.4.3, "Screening of Single Module Accident Sequences on Core Damage Frequency," the COL applicant would first screen external event sequences with an initiating event frequency less than 1E-5 per year. In LTR Table 1-2, "Definitions," an external event is defined as a hazard originating outside a nuclear power plant that directly or indirectly causes an initiating event and may cause safety system failures or operator errors that may lead to core damage or large early release. Internal fires from sources inside and outside the plant are also considered to be external events.

Based on the guidance and expectations outlined above, the staff would expect external and internal events to be screened based on equivalent screening criterion consistent with the 1995 PRA policy statement, SECY 04-0118, and RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3 (ADAMS Accession No. ML17317A256).

Request

Therefore, the staff requests that the applicant justify in the LTR:

(a) the technical basis for screening external events using a different quantitative screening criterion from internal events given that the ASME/ANS PRA standard addresses internal flood, internal fire, seismic, wind, external flood and other external hazards.

(b) why the methodology is not screening potentially risk significant external events given that external event initiators with a frequency less than 1E-5 per year could have core damage sequence frequencies that result in the QHO's being exceeded.

For example, an external event with an initiating event frequency of 5E-6 per year that causes core damage 50 percent of the time, has a resulting core damage sequence frequency of 2.5E-6 per year. The staff understands that the methodology in the LTR would allow this sequence to be screened out using the screening for external event initiators although it may challenge the large release frequency (LRF) Commission goal for new reactors and, consequently, the QHOs. As another example, regarding the external event screening initiator frequency of less than 1E-5 per year, in RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1 (ADAMS Accession No. ML070360253), the design-basis tornado is based on an annual exceedance frequency of 1E-7 per year, and, therefore, is not screened from deterministic analysis.

(c) As an alternative to (a) and (b) above, the applicant may provide revised external event screening criteria with technical justification.

01.05-45

Issue

The 1995 PRA policy statement, Draft Guide 1350, "Emergency Planning for Small Modular Reactors (SMRs) and Non Light Water Reactors (ANLWRs)," (ADAMS Accession No.ML18082A044) and NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," Revision 1, Final Report (ADAMS Accession No. ML17062A466), provide the expectation and guidance for the treatment of uncertainties in a risk-informed application.

Request

The staff requests that the applicant address in the LTR how numerical uncertainties associated with each screened core damage sequence are to be considered when comparing against the numerical screening thresholds. Consistent with NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," Revision 1 (ADAMS Accession No. ML17062A466), the applicant should address the following:

(a) The impact of parameter uncertainty on the screening results.

(b) A description of the relevant sources of model uncertainty and their impact on the screening results.

(c) A description of any significant modeling assumptions and their impact on the screening results.

The treatment of uncertainty in the LTR should consider that a COL applicant will not have operating procedures, operating experience (especially for new design features), or the ability to perform walkdowns.

01.05-46

Issue

DC/COL ISG-028, "Assessing the Technical Adequacy of the Advanced Light-Water Reactor Probabilistic Risk Assessment for the Design Certification Application and Combined License Application," (ADAMS Accession No. ML16130A468), along with RG 1.200, " An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 3 (ADAMS Accession No. ML20238B871), provides an acceptable approach to support certification and licensing of reactors under 10 CFR Part 52. It further states that other applications, including risk-informed applications, "need to directly address the application-specific regulations and guidance, including the evaluation of the technical adequacy of the PRA needed for the specific application using the PRA Standard, as endorsed by RG 1.200."

Request

Therefore, the staff is requesting that the applicant address in the LTR:

(a) The need for PRA to be peer reviewed in accordance with NEI 17-07, Revision 2 (for Advanced Light Water Reactors (ALWRs)) (ADAMS Accession No. ML19231A182).

(b) The need for the COL applicant to evaluate hazards/modes where NRC-endorsed Standards do not exist to justify the technical adequacy of the PRA to support the PRA sequence screening.

(c) The need for the PRA to be developed using RG 1.200 for Capability Category II.

(d) The need for the COL applicant to identify and justify any exceptions (e.g., inability to perform walkdowns).

01.05-47

Issue

Both the 2009 version of the Level 1/LERF LWR PRA standard (ASME/ANS RA-Sa-2009) and the current draft of the next edition of the Level 1/LERF LWR PRA standard (ASME/ANS RA-S-1.1) include the terms significant accident sequence, significant accident progression sequence, significant cut set and the definitions thereof. The definitions of those terms have built into them quantitative criteria related to when something is considered to be significant and, therefore, need to be considered (i.e., cannot be dismissed). Specifically, the criteria classify a contributor as significant when they are part of the top 95 percent of the total risk or any individual contributor is more than 1 percent of the total risk. According to the LTR, Section 3.4.3, "Screening of Single Module Accident Sequences on Core Damage Frequency," all significant internal and external accident sequences can be screened if they have a core damage frequency less than 1E-7 per year. The staff is concerned that parsing of core damage sequences into individual components for comparison against numerical screening thresholds could screen out potentially risk significant core damage sequences.

Request

For ALWRs, the staff is requesting that the LTR be revised to address the concern of parsing sequences to limit parsing and to ensure consistent application of the methodology.

01.05-48

Issue

The staff did not find information in the LTR about potential releases due to non-core damage events that would necessitate protective actions.

Request

The staff is requesting that the LTR include potential releases due to non-core damage events that would necessitate protective actions consistent with the Environmental Protection Agency (EPA) Protective Action Guidance (PAGs). For example, dropping the upper portions of the NuScale reactor pressure vessel and the containment vessel as they are moved to or from the dry dock area, onto the fuel in the lower Reactor Pressure Vessel (RPV), which remains in the refueling flange tool may cause mechanical fuel damage and a gap release.