RAIO-0817-55464

August 16, 2017

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 69 (eRAI No. 8809) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 69 (eRAI No. 8809)," dated June 21, 2017

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosure to this letter contains NuScale's response to the following RAI Questions from NRC eRAI No. 8809:

- 19-9
- 19-10
- 19-11
- 19-12

This letter and the enclosed response make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8809

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8809

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8809
Date of RAI Issue: 06/21/2017

NRC Question No.: 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.

NuScale Response:

Part 1: With regard to the screening criteria for SSCs:

Based on insights summarized in the following paragraphs, there are no nonsafety-related SSCs that contribute to potential initiating events included with the RTNSS program:

a. The calculation of initiating event frequencies considers nonsafety-related SSCs. Initiators considered in the internal events, full power PRA are discussed in FSAR Section 19.1.4.1.1.2; these initiators are also considered in the low power and shutdown (LPSD) PRA as described in FSAR Section 19.1.6.1.1. The following full power initiators consider nonsafety-related SSCs:
    - chemical and volume control system (CVCS) pipe breaks outside of containment (IE-CVCS--ALOCA-COC, IE-CVCS--ALOCA-LOC),
    - secondary side line breaks (IE-TGS---FMSLB-UD),
    - loss of offsite power (IE-EHVS—LOOP),
    - loss of DC power (IE-EDSS—LODC),
    - general reactor trip (IE-TGS---TRAN-NPC), and
    - loss of support systems (IE-TGS---TRAN-NSS).

   As stated in the response to Part ‘B’ of this question, LPSD initiators include those considered in the full power PRA and also a module drop in the operating area or refueling area.
b. Unavailability of nonsafety-related SSCs would either (i) preclude module operation (e.g., CVCS, feedwater, or DC power), which indicates that such components do not contribute

to the frequency of an initiating event or (ii) require that another nonsafety-related SSC (e.g., an AC bus) be aligned to support module operation, which indicates that unavailability of these SSCs would have little effect on the initiating event frequency. As such, unavailability of a nonsafety-related SSC does not significantly affect the calculation of the initiating event frequencies.

c. In applying the screening criteria to determine whether nonsafety-related SSCs could have a significant effect on an initiating event, NuScale applied a threshold of 20 percent based on the the methodology discussed in approved Licensing Topical Report TR-0515-13952-A “Risk Significance Determination”. As discussed in FSAR Section 19.1.4.1.1.9, the criteria were developed to support the relatively simple, passive NuScale design with risk estimates that are significantly lower than those of operating plants. Further, Section 3.1.3 of TR-0515-13952-A describes that the Fussell-Vesely (FV) metric of 20 percent also applies to internal initiating events. As such, in determining whether an initiating event significantly affects the CDF and the LRF, internal event initiators that contribute 20 percent or more to risk were evaluated for potential risk significance. Because the 20 percent threshold used for initiating events is the same as that used for SSCs in the NuScale design, using this value is judged to be conservative in comparison to the typical approach in which there is a factor of twenty difference between the thresholds for initiating events and SSCs based on generic guidance (i.e., 10 percent threshold for initiating events, provided in SRP 19.3 Areas of Review 5.c, compared to a 0.5 percent threshold for SSCs provided in RG 1.200).

   As identified in FSAR Tables 19.1-20, 19.1-27, and 19.1-70, initiating events were included in the list of candidates for risk significance based on the PRA, but evaluated as unnecessary for inclusion in the RTNSS program. FSAR Section 19.3.4 has been updated to clarify that initiating events were considered in the evaluation of nonsafety-related SSCs to be included in the RTNSS program.

<u>Part B: With regard to evaluation of nonsafety-related SSCs that could prevent module drop:</u>

As described in FSAR Section 9.1.5, the principal components of the overhead heavy load handling systems (OHLHS) used for refueling are the reactor building crane (RBC), the module lifting adapter (MLA), and associated hoists and load handling devices; these components are not safety-related. The RBC and MLA are used to lift and transport modules for refueling. The RBC is a bridge crane with a trolley and main hoist; the MLA provides the connection method for the RBC to lift and carry a module from the operating bay to the refueling pool and dry dock. The RBC is designed to the Type 1 crane requirements in ASME NOG-1, which satisfy the single-failure-proof requirements in NUREG-0554. Similarly, the MLA is designed as a single-failure-proof lifting device in accordance with ANSI N14.6 and ASME NOG-1. As such, the failure of any single nonsafety-related component does not result in a module drop. The RBC and MLA are also protected from the effects of external missile hazards by being located inside the Seismic Category I reactor building. As identified in FSAR Table 17.4-1, the RBC and MLA are classified as risk-significant.

In the event of a loss of power, the RBC and brakes will set to stop all motion. The RBC is designed with redundant holding brakes; if one set fails to engage, the other brake automatically holds the load. Both brake systems are designed for the maximum allowable crane load. Since a loss of electric power does not does result in a module drop, power-related SSCs are not included in the scope of the RTNSS program.

The RBC bridge and trolley have redundant position control systems. Failures or a loss of power will result in stopping the control system and automatically setting the crane braking system. Because control system failures do not result in a module drop, none were identified in the scope of the RTNSS program.

As described in FSAR Section 19.1.6.1, the RBC moves a module from its operating bay to the refueling pool where it is set in the containment flange tool (CFT). The CFT is a support stand that is mounted to the bottom of the refueling pool and provides a support for the lower containment vessel (CNV) during module disassembly. The module remains attached to and supported by the crane when in the CFT; therefore, the CFT is not relied on to prevent module drop. After the CNV is disassembled, the reactor pressure vessel (RPV) and upper CNV are lifted out of the CFT and moved to the reactor flange tool (RFT). The RFT is a support stand that is mounted to the bottom of the refueling pool. Because the RPV and upper CNV remain attached to the RBC when the RPV is assembled in the RFT, the RFT is not relied on to prevent a module drop. Therefore, because the CFT and RFT do not support the module independent of the RBC, they do not contribute to the initiating event frequency.

Impact on DCA:

Section 19.3.4 has been revised as described in the response above and as shown in the markup provided in this response.

identification of the active nonsafety-related functions that interface with the passive safety-related functions. The interactions between the systems were then analyzed to identify potential adverse interactions that could preclude the passive safety-related functionality from being accomplished. This systematic evaluation did not identify any significant adverse interactions between the active nonsafety-related systems and the passive safety-related systems. Therefore, no nonsafety-related SSC meet the RTNSS E criteria.

19.3.3 Functional Design of RTNSS Structures, Systems, and Components

A reliability/availability (R/A) mission is a set of requirements related to the performance, reliability, and availability of a risk-significant SSC function that adequately ensures the accomplishment of its task, as defined by the focused PRA or deterministic analysis.

No R/A missions are established for the nonsafety-related, risk-significant SSC since, as discussed in previous sections, no SSC are determined to meet the RTNSS criteria, and therefore, no RTNSS SSC are identified.

19.3.4 Focused Probabilistic Risk Assessment

The focused PRA is described in Section 19.1.9.3. The focused PRA is developed from the baseline PRA by removing nonsafety-related functions and their support from the baseline PRA model in order to assess the capability of the safety-related passive systems. The focused PRA demonstrates that credit for availability of nonsafety-related components is not needed to meet the Commission's CDF and LRF safety goals. Because the calculated risk metrics are much lower than the safety goals, risk and availability objectives are not established for nonsafety-related components.

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The focused PRA maintains the same scope of initiating events and their frequencies as identified in the baseline PRA. The initiating event frequencies developed in Section 19.1 include consideration of nonsafety-related SSC as event initiators. The full power and shutdown PRA models are reviewed to determine whether nonsafety-related SSC could have a significant effect on the estimated frequency of initiating events using the screening criteria:

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a) Does the calculation of the initiating event frequency consider the nonsafety-related SSC?

b) Does the unavailability of the nonsafety-related SSC significantly affect the calculation of the initiating event frequency?

c) Does the initiating event significantly affect the CDF and the LRF?

RAI 19-9

Based on the NuScale risk significance criteria discussed in Section 19.1.4.1.1.9, internal event initiators that contribute 20 percent or more to risk are evaluated for potential risk significance. Nonsafety-related SSC that contribute to potential initiating events are evaluated as unnecessary for inclusion in the RTNSS program because unavailability of nonsafety-related SSC would either (i) preclude module operation (e.g., CVCS), such that it would no longer contribute to an initiating event frequency or (ii) require that another nonsafety-related SSC (e.g., an AC bus) be aligned to support module operation, which indicates that unavailability has little effect on the initiating event frequency. The initiating event frequencies are generally based on generic industry data as discussed in Section 19.1.4.1.1.5. Additionally, sensitivity studies in Table 19.1-22 indicate that the CDF and LRF for the baseline PRA are not sensitive to initiating event frequencies.

The results of the focused PRA are considered in the development of the technical specification requirements. No nonsafety-related design features or functions are relied on to reduce the CDF or LRF below the Nuclear Regulatory Commission goals.

As discussed earlier, the focused PRA supports the identification of RTNSS C and RTNSS D SSC, while contributing to identifying RTNSS B SSC. No RTNSS B, RTNSS C, or RTNSS D SSC have been identified for the NuScale Power Plant design as a result of insights from the focused PRA.

19.3.5 Augmented Design Standards

Augmented design standards are required for RTNSS B SSC to assure reliable performance in the event of applicable hazards, such as natural phenomena. These natural phenomena hazards include safe shutdown earthquake, hurricane and tornado winds, and floods including internal flooding.

RTNSS B SSC are also required to be designed such that safety functions required in the post 72-hour through 4-day period following an accident can be accomplished with the required onsite equipment and supplies.

Since no RTNSS B SSC are identified for the NuScale Power Plant design, no RTNSS augmented design standards are applied.

19.3.6 Regulatory Treatment of RTNSS SSC

Regulatory oversight of RTNSS SSC may include Maintenance Rule (monitoring the effectiveness of maintenance), and either the Technical Specifications or a licensee controlled Availability Controls Manual.

The Availability Controls Manual is established in a manner similar to Technical Specifications and includes availability control limited conditions of operation (ACLCO) and availability controls surveillance requirements. Availability controls are commensurate with the assumptions in the PRA, and include, at a minimum, RTNSS B SSC. The establishment of ACLCO and surveillance requirements provides assurance that the RTNSS SSC can meet their R/A missions and that the component availability is consistent with its R/A mission.

Since no RTNSS SSC are identified, no additional regulatory oversight is required for any nonsafety-related risk-significant SSC.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8809
Date of RAI Issue: 06/21/2017

NRC Question No.: 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).

NuScale Response:

FSAR Section 19.1.5.5.1 includes a reference to Table 19.1-61 with the key modeling assumptions for the high-winds PRA; it includes the assumption that forecasting information (e.g., hurricane warnings) will be used and mitigative actions (to delay startup or move a module from the crane to a safe position) taken when forecasts or conditions indicate a high winds impact is likely. This assumption is maintained by COL Item 19.1-8, which states:

> "A COL applicant that references the NuScale Power Plant design certification will confirm the applicability of assumptions and data and modify as necessary for the as-built/as-operated probabilistic risk assessment."

A COL applicant that incorporates the NuScale design certification but, in the site-specific PRA, does not conform to the design certification assumptions would be departing from the design and analysis described in the DC FSAR and would be required to identify and justify the departure, as required by regulations in 10 CFR 52.79(d)(1). The conformance of individual COL applicants to the design certification and the NRC regulations are confirmed during the review of the COL application and issuance of the individual license.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

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eRAI No.: 8809
Date of RAI Issue: 06/21/2017

NRC Question No.: 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 DCCOL 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. DCCOL 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.

NuScale Response:

1. As described in FSAR Section 2.4.2, the design assumes the maximum flood elevation (including wind-induced wave run-up) is one foot below baseline plant elevation; the baseline plant elevation is the top of the concrete on the ground floor of the reactor building. A second, related, key design parameter is the assumption that the site is properly graded and has adequate drainage to prevent localized flooding from the maximum precipitation event; these design parameters are maintained by COL Item 2.4-1 in recognition that the potential for external flooding is site-specific.

   As discussed in Part 8 of the ASME/ANS Standard (FSAR reference 19.1-1), external-flooding risks are generally not found to be important contributors to overall risk at nuclear power plants because of siting requirements. In addition, most large external floods occur over a long enough duration to allow the plant operating staff to take appropriate steps to secure the plant and its safety-related SSCs. However, to develop a probabilistic risk assessment of a potential external flood, an external flood with a recurrence interval of one in five hundred years was assumed to bound the likelihood of exceeding the design basis flood. This assumption is identified in Table 19.1-58, and maintained by COL Item 19.1-8.

2. The following excerpt from Part 8 of the ASME/ANS Standard is the basis for the assumption that operators are assumed to cease refueling and crane operations, and perform a controlled shutdown prior to potential equipment impacts, in 90 percent of external floods that exceed the design basis:

   > "… external-flooding risks are generally not found to be important contributors to overall risk at nuclear power plants. One major reason is that the siting requirements are intended to assure this outcome, and by and large they have been successful in that ... Another key reason is that most large external floods occur only after significant warning time or over a long enough duration to allow the plant operating staff to take appropriate steps to secure the plant and its safety-related SSCs. The PRA team is therefore urged to take as much credit for warning time and compensatory actions as the plant's planning and procedures allow …."

This assumption is identified in Table 19.1-58 and maintained by COL Item 19.1-8.

3. FSAR Section 19.1.5.4 describes the susceptibility of buildings from an external flood. Equipment located inside the containment vessel (CNV) is designed to operate in harsh environments; therefore, flooding effects are not considered for equipment inside the CNV. Both the control building and reactor building are judged susceptible to an external flood that exceeds the design basis. Nonsafety-related and electrical equipment are assumed to be susceptible to flood damage; safety-related components move to their fail-safe position and nonsafety equipment is unavailable. Flood protection features are not credited in the external flooding analysis; thus, there are no flooding penetrations that are identified as risk significant. FSAR Section 19.1.5.4.2 (Risk Significance) will be updated to reflect this modeling approach.

Impact on DCA:

Section 19.1.5.4.2 has been revised as described in the response above and as shown in the markup provided in this response.

majority of external flood risk is associated with flood-induced LOOP followed by an incomplete ECCS actuation. Incomplete ECCS actuations are the result of CCFs of both RRVs or all three RVVs following the flood-induced failures of the switchgear and batteries. The most significant cutsets for LRF also involve CCFs of the containment evacuation system CIVs or chemical and volume control system CIVs to close.

Risk Significance

Applying the methodology referenced in Section 19.1.4.1.2, the results of the external flooding PRA were reviewed to identify candidate risk significant SSC. Based on the review, the following SSC are risk significant, as summarized in Table 19.1-64:

- CVCS and CES isolation valves
- ECCS
- MPS

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For the situations in which a controlled shutdown has not been completed by the operators prior to external flooding (i.e., ten percent of the external flooding initiating events) no operator actions are credited. Therefore, no operator actions are identified as risk significant in the external flooding evaluation. In addition, because flooding penetrations (e.g., doors) are not credited in the external flooding analysis, none were identified as risk significant.

Key Assumptions

Table 19.1-58 summarizes the key assumptions associated with the external flooding PRA.

Uncertainties

Parameter uncertainty associated with the external flooding PRA is characterized by probability distributions associated with the calculated results. Section 19.1.4.1.2 identifies sources of uncertainty in the Level 1 internal events model and Section 19.1.4.2.2 identifies sources of uncertainty in the Level 2 internal events model. Model uncertainties that are unique to the external flooding PRA include the external flooding initiating event frequency and the lack of design detail on protective and mitigative features. Model uncertainties associated with the external flooding PRA are addressed with assumptions or sensitivity studies as indicated in the following sections.

Sensitivity Studies

The external flooding PRA for design certification is based on conservative, bounding assumptions. No sensitivity studies were performed.

Key Insights

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8809
Date of RAI Issue: 06/21/2017

NRC Question No.: 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.

NuScale Response:

Technical Report TR-1116-52011 addresses risk associated with each operating phase including low power and shutdown activities, which includes refueling activities, as well as external event risk. This is illustrated by the inclusion in the NuScale Technical Specifications of limiting condition of operation (LCO) 3.8.1, which specifies requirements for nuclear instrumentation during refueling operations.

The risk consideration for each operating phase, including external event hazards conditions, is described in FSAR Section 19.1.4.1.1.9, and the results are presented in

- FSAR Table 19.1-20: "Listing of Candidate Risk Significant Structures, Systems, and Components (Full Power, Single Module) Level 1 Probabilistic Risk Assessment",
- FSAR Table 19.1-27: "Listing of Candidate Risk Significant Structures, Systems, and Components (Full Power, Single Module) Level 2 Probabilistic Risk Assessment",
- FSAR Table 19.1-64: "Listing of Candidate Risk Significant Structures, Systems, and Components: External Events", and
- FSAR Table 19.1-70: "Listing of Candidate Risk Significant Structures, Systems, and Components (Single Module): Low Power and Shutdown Probabilistic Risk Assessment".

The SSCs identified by the PRA in these tables were further evaluated under the D-RAP process, as discussed in FSAR Section 17.4.5. With two exceptions, each system and component listed in the tables, specifically including Tables 19.1-64 and 19.1-70, is addressed in the Technical Specifications because they met other 10 CFR 50.36 criteria for inclusion.

The two exceptions from inclusion in the Technical Specifications are the combustion turbine generator (CTG) listed in Table 19.1-20, and the reactor building crane (RBC), listed in Table 19.1-70.

The CTG is not included in the Technical Specifications because the CTG is identified as risk significant only for postulated internal events accident sequences which have an exceedingly low calculated CDF, as illustrated in Table 19.1-80. Further, the PRA used conservative modeling assumptions when considering this backup power supply, as described in FSAR Table 19.1-21:

> The electric power system fault trees include modeling simplifications (e.g., automatic cross-ties are not modeled, power supplies to shared systems are modeled as though they are provided exclusively from module 1.)

The RBC is a risk significant component as indicated in FSAR Table 17.4-1. However it is not included in the Technical Specification because it does not meet the immediacy consideration for inclusion as described in the NRC *Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors*, 58 FR 39132, and the issuance of 10 CFR 50.36, in 60 FR 36953. Reactor building crane operation is a deliberate and carefully implemented evolution. The crane design, maintenance, inspection, and operations procedural controls assure that emergent conditions will not create an immediate threat to the health and safety of the public during its use.

Management of risk associated with the RBC is by design, and programmatic controls, consistent with those applied in similar situations in the industry.

Although presented in FSAR Table 19.1-70 as a single component, the RBC is composed of numerous subcomponents, which include special lifting devices. The RBC function includes relevant subcomponents and the RBC is designed to be single-failure proof as described in FSAR Section 9.1.5. Maintenance, inspection, and testing of the RBC and its subcomponents are prescribed by ASME B30.2, as stated in FSAR Section 9.1.5.4. These types of activities are more suitable for programmatic implementation and are typically implemented as such in the commercial nuclear industry. The requirements for defining and performing these control activities have been relocated to operating programs and procedures, and are established in FSAR Section 9.1.5. These considerations are controlled in accordance with 10 CFR 50.59, consistent with the NRC *Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors*. In addition to the design and controls established by the FSAR,

- COL Item 9.1-5 directs a COL applicant to describe the process for handling and receipt of critical loads including NuScale Power Modules, and
- COL Item 13.5-3 directs a COL applicant to describe the program for developing maintenance and operating procedures which will include implementation of the requirements in FSAR Section 9.1.5.

Based on these considerations, the RBC was not included in the Technical Specifications.

As detailed above, Technical Report TR-1116-52011 included consideration of risk in each operating phase as well as external event risk. One LCO was selected for inclusion that is specifically applicable during refueling operations based on operating experience. Application of the PRA selection criteria as described in 10 CFR 50.36, including the NRC considerations and policy associated with the regulation, did not identify additional systems or components for inclusion. Further, because multi-module CDF is almost a factor of ten lower than the single module CDF, and module-specific equipment failure is a precursor to core damage, multi-module considerations did not result in additional Technical Specification requirements.

Section 19.1.8.6 of the FSAR has been modified to link the PRA Technical Specification input to the discussion of Technical Specifications in Chapter 16.

Impact on DCA:

FSAR Section 19.1.8.6 has been revised as described in the response above and as shown in the markup provided in this response.

design certification stage to support development of the Design Reliability Assessment Program, as discussed in Section 17.4.

19.1.8.5 Probabilistic Risk Assessment Input to the Regulatory Treatment of Nonsafety-Related Systems Program

The PRA is used to support the identification of non-safety related SSC that are within the RTNSS scope at the design certification stage. The scope, criteria and process to determine SSC within the RTNSS program are discussed in Section 19.3.

19.1.8.6 Probabilistic Risk Assessment Input to the Technical Specifications

The PRA provides input to the technical specifications from several perspectives:

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- Criterion 4 of 10 CFR 50.36(c)(2)(ii)(D) requires that a limiting condition of operation be established for an SSC which operating experience or PRA has shown to be significant to public health and safety. The design certification PRA is used to identify SSC meeting this criterion by applying the quantitative criteria discussed in Section 19.1.4.1.1.9. (See Section 16.1.1.)
- Surveillance frequencies in the technical specifications are consistent with assumptions made in the design certification PRA.
- The PRA may be used to support development of Risk Managed Technical Specifications, as described by NEI 06-09 (Reference 19.1-5).
- The PRA may be used to support development of a Surveillance Frequency Control Program as described by NEI 04-10 (Reference 19.1-4).

19.1.9 Conclusions and Findings

Key insights from the Level 1 and Level 2 PRA for internal events and external events, full-power and LPSD modes, as well as single and multiple module operation were provided in earlier sections. The analysis demonstrates that the NuScale Power Plant design incorporates features that produce an exceedingly low risk to public health and safety. Key results of the analysis and additional risk perspectives are provided in this section, specifically:

- conformance with safety goals
- perspective of the NuScale small core with respect to safety goals
- focused PRA insights
- unique system capability

19.1.9.1 Conformance with Safety Goals

The safety goal policy statement and subsequent guidance provide quantitative objectives for evaluating conformance with the qualitative goals associated with public health and safety. The quantitative results of the PRA, summarized in Table 19.1-80, demonstrate that the risk associated with operation of a NuScale Power Module is substantially less than defined by the safety goals. The table also indicates that