U.S. NUCLEAR REGULATORY COMMISSION SUMMARY OF THE SEPTEMBER 13, 2022, OBSERVATION PREAPPLICATION PUBLIC MEETING WITH SMR, LLC (A HOLTEC INTERNATIONAL COMPANY) TO DISCUSS CONTROL ROD DRIVE SYSTEM TO SUPPORT THE CONSTRUCTION PERMIT APPLICATION OF THE SMR-160 DESIGN

Meeting Summary

The U.S. Nuclear Regulatory Commission (NRC) held an observation preapplication public meeting on September 13, 2022, with SMR, LLC (SMR), a Holtec International Company, to discuss questions regarding the control rod drive system (CRDS) to support the construction permit application for the SMR-160 design. Specifically, SMR requested the meeting to provide the NRC staff with a high-level overview of the SMR-160 CRDS design and the associated NRC Standard Review Plan (SRP) Section 3.9.4 testing provisions and to discuss acceptable analytical methods that can be used in lieu of prototype testing, and industry experience and plant referenced designs that preclude the use of analytical methods.[1,2]

This virtual meeting included attendees from the SMR, LLC, Holtec International, LLC, and NRC staff. There were no members of the public observing the meeting.

- The applicant provided an overview of the SMR-160 CRDS noting that it is based on designs in operation at various pressurized-water reactors (PWRs). The applicant discussed its process of reviewing different reactor designs and associated quality assurance programs to determine compliance with SRP Section 3.9.4 and to develop the SMR-160 CRDS. The applicant detailed the results of its review for the NuScale, AP1000, and APR1400 plant designs.

- The NRC noted that a designer is the most knowledgeable resource on the specifics of its particular reactor design. In reviewing a reactor application to satisfy SRP Section 3.9.4, the NRC staff focuses on the ability of the CRDS to shut down the reactor and keep it safely shut down under all postulated scenarios over the life of the plant. A comparison to previous reviews can be helpful to an applicant in understanding how the NRC approaches the review process and how certain issues may be more significant than others. The NuScale Design Certification (DC) Final Safety Analysis Report (FSAR) identified unique design features, such as a longer control drive shaft and a remote disconnect mechanism and discussed its operability assurance program.[3] The NRC Safety Evaluation Report (SER) for the NuScale DC and the NuScale FSAR may be

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1 SMR, LLC, "Preapplication Materials for September 13, 2022 (Project No. 99902049," dated September 13, 2022. Agencywide Documents and Access Management System (ADAMS) Accession No. ML22256A019.

2 U.S. NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (NUREG-0800, Formerly issued as NUREG-75/087)," Chapter 3, "Design of Structures, Components, Equipment, and Systems," Section 3.9.4, "Control Rod Drive Systems," ML16133A472.

3 NuScale, "NuScale Standard Plant Design Certification Application," Chapter 3, "Design of Structures, Systems, Components and Equipment," ML20225A154.

useful references for preparation of an application, as it demonstrates a previously successful process for the review of an application.[4]

- In response to the applicant's question on what is the NRC definition of "new design," the NRC staff noted that the NRC regulations in 10 CFR 50.43, "Additional standards and provisions affecting class 103 licenses and certifications for commercial power," specify requirements in paragraph (e) for safety features in nuclear reactor designs which differ significantly from light-water reactor designs that were licensed before 1997, or use simplified, inherent, passive, or other innovative means to accomplish their safety functions. In a traditional CRDS, electromagnetic coils and latches are used to step the control rods in and out of the reactor core of a PWR, including the NuScale design. If a design were to use a different means of moving the control rods, this would likely be considered a new CRDS design.

  Should an existing design be used as the basis for a modified design (for example, a smaller version of an existing design), then a correlation to other similar designs could be proposed by an applicant and considered in the NRC staff's review.

  The applicant noted that its CRDS design has many features identical to existing designs and noted minor variations or geometry differences. The applicant pointed to prior applications such as the NuScale, AP1000, and APR1400 as similar instances.

- For the question on NRC's definition of "new and unique features," the NRC staff referred to the NuScale application, which identified a longer control rod drive shaft and a remote disconnect mechanism. The NRC staff considered these features to be new and unique. An applicant could describe the safety features in a planned reactor design, and the NRC staff could evaluate whether the safety features would be considered new and unique.

  The applicant noted some of the features of the SMR-160 CRDS design that may be considered new and unique. The staff indicated that additional future discussion may be beneficial regarding these features.

- With respect to the NRC's definition of operability assurance as it relates to the overall control rod drive mechanism and CRDS function, the NRC staff responded that the reactor should be able to be shut down and safely kept shut down in all postulated conditions over the entire life of the plant. SRP Section 3.9.4 for the CRDS provides details regarding wear, functioning times, latching, and the ability to overcome a stuck control rod.

  The NRC staff noted that the other review areas associated with the CRDS will be discussed during the October 18, 2022, public meeting.

- In evaluating design stress limits for a new testing program, including fatigue limits and deformation limits in support of performance testing, stability testing, and endurance testing, the applicant requested what can be evaluated empirically versus analytically. The NRC staff responded that American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code) design reports have been referenced

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[4] U.S. NRC, "NuScale Design Certification Final Safety Evaluation Report," Chapter 3, "Design of Structures, Systems, Components and Equipment," ML20205L405.

successfully in past reviews to assure pressure housing components will perform acceptably in terms of stress limits. Correlation to existing empirical results may be possible, but it might depend on the rigor of the analysis. For example, NuScale conducted a prototype testing program of the CRDS for rod drop and shaft misalignment as audited and documented in Section 3.9.4.4.5 of the NuScale SER.[4]

The applicant requested clarification on whether the ASME BPV Code is referenced for the pressure retraining aspects of the design versus internal function or alignment for a dropped control rod. The NRC staff confirmed that the ASME BPV Code is for maintaining pressure retention and that functionality of the system would be addressed in other ways.

In response to the applicant's request on the regulatory risk of following an existing magnetic latch design with minor variations and relying on existing operating experience, the NRC staff responded that it is open to a future discussion on the design differences including identification of the differences and any potential effects.