U.S. NUCLEAR REGULATORY COMMISSION SUMMARY OF THE FEBRUARY 18, 2025, OBSERVATION PREAPPLICATION MEETING WITH SMR, LLC (A HOLTEC INTERNATIONAL COMPANY) TO DISCUSS THE SMR-300 ACCIDENT RADIOLOGICAL CONSEQUENCES METHODOLOGY TOPICAL REPORT

Meeting Summary

The following summarizes the discussion during the meeting:

- Following the U.S. Nuclear Regulatory Commission (NRC) staff's opening remarks and introductions, SMR (Holtec) began its presentation with the meeting agenda, purpose, and desired outcome. The purpose was to present a high-level overview of an upcoming accident radiological consequences methodology topical report (TR) and applicable regulatory guidance. The methodology overview focused on the calculation of radiation doses for the exclusion area boundary (EAB) and low population zone (LPZ) boundary determinations, calculation of radiation doses to the main control room and technical support center, and meeting the intent of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.34(a)(1)(ii)(D)[1] and 10 CFR 50.34(b)(11). The desired outcome was to obtain feedback from the NRC staff on the approach for complying with the regulations.

- SMR (Holtec) described source terms of accidental release from containment models. Release fractions include chemical form fractions of iodine adopted from assumption A-1.1 of Regulatory Guide (RG) 1.183, Revision 1[2], which typically apply for emergency core cooling systems (ECCS) containment sump fluid pH ≥ 7. SMR-300 is designed to maintain primary coolant water pH ≥ 6. SMR (Holtec) referred to Figure 3.1 of NUREG/CR-5950[3] to demonstrate that chemical form fractions as described in RG 1.183, Revision 1 are also applicable to the SMR-300 design. The NRC staff acknowledged the pH deviation, confirmed its familiarity with the models of NUREG/CR-5950, and offered no objections, as long as SMR (Holtec) is able to justify why the chemical form fractions models in RG 1.183, Revision 1, are applicable to the SMR-300 design.

- SMR (Holtec) confirmed that the NRC's computer code RADTRAD version 5.03, is being used to assess the design basis accident (DBA) radiological consequences.

- SMR (Holtec) pointed out that SMR-300 control room habitability is not a safety-related function as control room operators serve no safety-related function as defined in 10 CFR 50.2," Definitions." SMR (Holtec) explained control room habitability is provided by two non-safety systems; control room ventilation (CRV) and breathing air and pressurization (BAP) systems. The NRC staff indicated that NuScale's control room design, which claims a non-safety-related function, is similar to what SMR (Holtec) described, and recommended that SMR (Holtec) review the NRC staff's safety evaluation report for the

[1] Title *10* of the *Code of Federal Regulations* (CFR), 10 CFR 50.34, "Contents of applications; technical information."

[2] U.S. NRC, Regulatory Guide (RG) 1.183, Revision 1, "Regulatory Guide (RG) 1.183, Revision 1, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," October 10, 2023. (ML23082A305)

[3] U.S. NRC, NUREG/CR-5950, "Iodine Evolution and pH Control," December 31, 1992. (ML063460464)

NuScale Design Certification Document.[4] SMR (Holtec) clarified that that the CRV system isolates itself when it senses high radiation setpoints at the CRV inlet. In addition, the CRV and BAP are separate systems with separate air pathways, which is a defense-in-depth feature.

- SMR (Holtec) clarified that it would develop radionuclide inventories for DBAs analyses consistent with the methods described in RG 1.183, Revision 1.

- SMR (Holtec) specified 25 rem for EAB and LPZ dose criteria for both the maximum hypothetical accident (MHA) loss of coolant accident (LOCA) and the “DBA LOCA”. Through dialog, the NRC staff understood that SMR (Holtec) was using the term “DBA LOCA” in reference to the analysis required by 10 CFR 50.46[5] to evaluate the acceptability of ECCS. To avoid confusion, the remainder of this summary will refer to this accident as the “10 CFR 50.46 LOCA.” The NRC staff pointed out that in RG 1.183, Revision 1, there is no acceptance dose criteria for the 10 CFR 50.46 LOCA because historically this event has not resulted in fuel failures. In a predecisional version of draft regulatory guide DG-1425[6], which is the precursor to Revision 2 of RG 1.183,, the NRC proposes a lower acceptance criterion of 6.3 rem for use in situations where the 10 CFR 50.46 LOCA analysis predicts small fuel failures (i.e., much smaller than the fuel failure considered in MHA analyses). This is intended to accommodate fuels with increased enrichments and higher burn ups, where traditional 10 CFR 50.46 LOCA analysis could calculate some fuel pin failures. SMR (Holtec) acknowledged the NRC's position and indicated it will review the predecisional DG-1425 acceptance criteria and provide more explanation in the TR, if necessary. However, the NRC staff offered that SMR (Holtec) could instead remove the 10 CFR 50.46 LOCA from the dose acceptance criteria in the TR in order to be consistent with RG 1.183, Revision 1.

- The NRC staff confirmed that the MHA LOCA and 10 CFR 50.46 LOCA are always considered distinct events and are analyzed separately. The NRC staff explained that the 10 CFR 50.46 LOCA analysis is intended to provide insight into the appropriate size of the ECCS based on a double-ended guillotine break assumption, and meeting the five criteria in 10 CFR 50.46(b). This is a defense-in-depth analysis where the size of the ECCS is set to prevent or limit fuel damage. However, for MHA LOCA, a substantial amount of core melt is deterministically assumed in order to evaluate the acceptability of the containment and fission product mitigation systems.

- SMR (Holtec) clarified that the SMR-300 will not operate with extended burn ups and will not use higher enrichments, and expressed concern that RG 1.83, Revision 2, guidance will not be published by the time the TR is submitted. The NRC staff offered that SMR (Holtec) could remove the 10 CFR 50.46 LOCA from the dose acceptance criteria in the TR in order to be consistent with RG 1.183, Revision 1. The NRC staff added that if the 10 CFR 50.46 LOCA analysis results in breaching the fuel barrier, then SMR (Holtec)

---

[4] U.S. NRC, “PHASE 6 - NuScale DC Final Safety Evaluation Report (Complete with Appendices),” August 28, 2020. (Package ML20023A318)

[5] 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.”

[6] U.S. NRC, Predecisional white paper, “DG-1425 (RG 1.183 Rev 2) Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors ACRS Version,” November 12, 2024. (ML24304A864)

would need to provide and justify a dose-based acceptance criteria similar to what the staff is proposing through DG-1425.

- SMR (Holtec) inquired if there is a concern among the NRC staff regarding an MHA LOCA event that assumes the full core damage in conformance with RG 1.183, Revision 1, for deterministic dose analysis for measuring the doses at the EAB and LPZ, and potentially having different assumptions for the emergency planning zone (EPZ) sizing methodology. The NRC staff explained that for SMRs, EPZ sizing is governed by 10 CFR 50.160.[7] For 50.160, applicant should determine the radiological releases from the facility that are evaluated in the radiological dose assessment to aid in the determination of the plume exposure pathway EPZ. In its safety analysis report, the applicant describes the licensing basis events relevant to the facility. The applicant should consider these licensing basis events as candidates for the development of the radiological releases. These licensing basis events may include both design-basis accidents and beyond-design basis events. Event likelihood may be used to determine whether the accident should be included in the range of accidents used in this analysis. For light-water reactor power reactors, the licensing basis events should include the design-basis events, design-basis accidents, and beyond-design-basis events evaluated in Chapter 15, “Transient and Accident Analysis,” and Chapter 19, “Severe Accidents,” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.” However, the staff acknowledges that 10 CFR 50.160 doesn’t apply to SMR-300 because this design does not meet the definition of an SMR in 10 CFR 50.2[8]. However, SMR (Holtec) indicated its intention to apply for an exemption to use the requirements in 10 CFR 50.160 in lieu of 10 CFR 50.47[9] and 10 CFR Part 50, Appendix E. The NRC staff discussed how different source terms are used for different purposes. The staff provided an example of an approved methodology for developing a MHA LOCA source term that meets the intent of the MHA LOCA core melt source term[10]. Also, the NRC staff advised SMR (Holtec) to review the approved NuScale’s TR on EPZ sizing[11], where NuScale provided its version establishing the technical basis for plume exposure emergency planning zones using several source terms, specific to its reactor design. Also, the NRC staff pointed to RG 1.242, Revision 0, Appendix B[12], provides some more recent guidance for EPZ sizing. SMR (Holtec) acknowledged and added it has high confidence that the existing MHA LOCA assumptions in RG 1.183, Revision 1, would be sufficient for determining EAB and LPZ boundaries.

- The NRC staff clarified that it is acceptable to use different accident source terms for different purposes in the licensing basis of a facility as long as they are properly justified and they satisfy the intent of the underlying regulatory requirement(s). Additionally, it is

---

[7] 10 CFR 50.160, “Emergency preparedness for small modular reactors, non-light-water reactors, and non-power production or utilization facilities.”

[8] 10 CFR 50.2, “Definitions.”

[9] 10 CFR 50.47, “Emergency plans.”

[10] NuScale, LLC, Submittal of Topical Report, “Accident Source Term Methedology,” TR-0915-17565-NP-A, Revision 4, February 2020 (ML20057G132)

[11] NuScale Power, LLC, “NuScale Power, LLC, Submittal of Topical Report "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones," TR-0915-17772-NP-A, Revision 3,” June 10, 2022. (ML22299A145)

[12] U.S. NRC, RG 1.242, Revision. 0, “Rulemaking - Final Rule - Regulatory Guide - 1.242, Revision 0 - Emergency Preparedness for Small Modular Reactors and Other New Technologies,” November 9, 2023 (ML23226A036).

acceptable for licensees to develop accident source terms that are more closely based on the particular designs being pursued. For example, historically, small modular reactor designers have established their own MHA release fractions specific to their designs, which are typically lower than the release fractions in RG 1.183, Revision 1, because they have much smaller reactors (compared to large light water reactors) with passive systems. Applicants have utilized either MELCOR computer code or the Modular Accident Analysis Program to estimate these source terms. The NRC staff added that SMR (Holtec) could use SMR-300 design specific release fractions for the MHA LOCA and could potentially follow a path similar to that of NuScale. SMR (Holtec) acknowledged and said they would likely also use MELCOR.

- The NRC staff pointed out that dispersion factor calculations for the EAB and the LPZ require the 99.5 percentile atmospheric dispersion coefficients so that the larger of the two χ/Q values, either the 99.5 percent maximum sector value or the 95 percent overall site value, is chosen to represent the χ/Q value for the 0–2 hour time interval, as recommended in RG 1.145, Revision 1[13], and RG 1.249, Revision 0.[14] SMR (Holtec) proposed using 95th percentile atmospheric dispersion coefficients, which are used for calculations for the control room. Since there is a departure from the atmospheric dispersion coefficient norm, the NRC staff stated they it expect SMR (Holtec) to provide a detailed discussion on this departure in the TR. SMR (Holtec) reasoned that because their EAB and LPZ boundaries are so close to the atmospheric dispersion release points, the 95th percentile atmospheric dispersion coefficient was appropriate. The NRC staff recapped that the MHA LOCA is the DBA that sets the design basis for the containment and the safety-related fission product mitigation systems consistent with 10 CFR 50.2 definition of safety-related structures, systems and components, criterion 3. For the MHA LOCA analysis, the facility designers deterministically assume a certain amount of core melt and determine how the containment and fission product mitigation systems function through dose-based acceptance criteria. The NRC staff observed that the deterministic source term that is provided in the NRC guidance for use in MHA LOCA analyses is developed by considering the results of several severe accidents; however, the MHA LOCA analysis itself is a DBA. Other DBA analyses are performed using different source terms – or, in some cases, no source term, to establish the design bases of different systems.

- The NRC staff clarified that SMR (Holtec) could remove the 10 CFR 50.46 LOCA dose-based acceptance criteria in the TR since they do not expect a challenge to the integrity of a fission product barrier.

- The NRC staff added that there are some large light-water reactors facilities that did their 10 CFR 50.46 LOCA analysis, with no fuel breach, and applied their RCS source term to perform the dose analysis. For these facilities’ safety analysis reports, American National Standards Institute Standard 18.1 is typically used to develop a normal operational RCS source term, which could be used as a source term in their dose analysis. This analysis tends to result in very low doses in the millirem range.

---

13 U.S. NRC, RG 1.145, Revision 1, “1983/02/28-Regulatory Guide 1.145, Revision 1 Atmospheric Dispersion for Potential Accident Consequence Assessments at Nuclear Power Plants,” February 28, 1983 (ML003740205)

14 U.S. NRC, RG 1.249, Revision 0, “RG 1.249 Rev 0 Use of ARCON Methodology For Calculation of Accident-Related Offsite Atmospheric Dispersion Factors,” August 9, 2023. (ML22024A241)

- SMR (Holtec) clarified that for radionuclide transport in the SMR-300 steam generator, which is a once-through steam generator similar to a Babcock and Wilcox type for modeling purposes, a break in the loop would result in the top void area becoming a release pathway. There would be no pre-scrubbing and once equilibrium is reached between the primary and secondary coolant systems, there should be no more steam flashing. This would be confirmed by a RELAP5 analysis. RG 1.183, Revision 1, commonly pertains to a U-tube steam generators with flashing. The NRC staff recommended that SMR (Hotlec) review the modeling of the NuScale design, which has a similar integrated design with a similar pressurizer and steam generator setup (albeit with a differing helical tube design). The NRC staff recommended providing precedence and references to approved similar designs. SMR (Holtec) acknowledged and committed to provide adequate justification for its methodology in the TR.

The closed session ended at 2:23 p.m.