U.S. NUCLEAR REGULATORY COMMISSION SUMMARY OF THE FEBRUARY 21, 2023, OBSERVATION PREAPPLICATION PUBLIC MEETING WITH SMR, LLC (A HOLTEC INTERNATIONAL COMPANY) TO DISCUSS THE SMR-160 CHAPTER 15 ANALYSIS

Meeting Summary

The U.S. Nuclear Regulatory Commission (NRC) held an observation public meeting on February 21, 2023, with SMR, LLC (SMR), a Holtec International Company (Holtec), to discuss preapplication information related to the SMR-160 Chapter 15 analysis.[1] Specifically, SMR (Holtec) requested the meeting to receive NRC staff feedback on the SMR-160 deterministic safety analysis codes and methods described in its presentation slides.[2, 3] The meeting was initially scheduled for February 15, 2023, and was rescheduled to February 21, 2023, due to conflicts.

This virtual observation preapplication meeting had attendees from SMR, LLC, (Holtec) the NRC staff, and members of the public. A closed session was conducted between the NRC staff and SMR (Holtec) to discuss proprietary information.

Preapplication engagements, including this meeting, provide an opportunity for the NRC staff to engage in early discussions with a prospective applicant to offer licensing guidance and to identify potential licensing issues early in the licensing process. No decisions or commitments were made during the preapplication meeting.

The following summarizes the discussion during the open session of the meeting:

- After opening remarks and introductions, SMR (Holtec) shared that the objective of the meeting was to obtain NRC staff feedback on its high-level overview of the computer codes and methods used to perform the transient and accident analyses for the SMR-160 design, and to identify specific topics for further discussion in future meetings.

- In reference to the computer code bases described on Slide 5, SMR (Holtec) noted the similarities of its design with existing nuclear power plants (NPPs) including materials and operating conditions. The NRC staff observed that the application should include additional information on how the codes will be validated considering the passive design of the SMR-160 and the smaller size of the reactor compared with existing NPPs.

  The NRC staff noted that in following the Evaluation Model Development and Assessment Process (EMDAP) in Regulatory Guide (RG) 1.203, the applicability and

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1 Letter from J. Hawkins, “SMR, LLC, Submittal of Preapplication Meeting Materials for February 15, 2023,” dated January 18, 2023, Agencywide Documents and Access Management System (ADAMS) Accession No. ML23018A010, part of ML23018A009.

2 SMR, LLC, “SMR, LLC, NRC Meeting: Deterministic Safety Analysis Codes and Methods Overview,” dated February 15, 2023, ML23018A012 - Public, part of ML23018A009.

3 SMR, LLC, “SMR, LLC, NRC Meeting: Deterministic Safety Analysis Codes and Methods Overview,” dated February 15, 2023, ML23018A011 – Proprietary, part of ML23018A009.

scalability of the computer codes to the design is an important factor because the design of several key structures, systems, and components and their interactions during normal operation, transients, and accidents are unique for SMR-160 design.[4]

- In response to the NRC staff's question on how the applicability and validation of the codes will be documented, SMR (Holtec) noted that it does not plan to submit topical reports as other applicants had done. Instead, SMR (Holtec) is considering the submission of a series of White Papers for NRC staff feedback and presenting the information in the application. The NRC staff encouraged SMR (Holtec) to consider the benefits previous applicants gained from having the documentation in a separate report from the application including minimal changes to the preliminary safety analysis report resulting from the review and ease of reference for other future applications.

- The NRC staff noted the lack of information on radiological release fractions since SCALE only provides decay heat and inventories. Holtec stated that they intend to use RG 1.183.[5]

- SMR (Holtec) confirmed that MELCOR Accident Consequence Code System and RADionuclide Transport, Removal and Dose Estimation are used in the SMR-160 probabilistic risk assessment.

- The NRC staff commented that RELAP5-3D is a good starting point and that the code should be modified based on integral testing to address specific design features. SMR (Holtec) responded that it plans to leverage integral and separate effect test results to fill in data sets.

- In response to the NRC staff's question on the considerations of how the computer codes are coupled, SMR (Holtec) responded that there have been some considerations and would request a future technical discussion on this topic.

- With respect to the code validation, the NRC staff observed that the neutron flux and source term distribution and dispersion are reactor system design dependent and codes used for these analyses should address applicability to the SMR-160 design. SMR (Holtec) stated it will request a future discussion on this topic.

- The NRC staff cautioned the use of code-to-code benchmarks without justification on applicability for a specific design. For example, if two codes are biased or trend in the same direction, the bias may not be identified by using code-to-code comparison. The computer codes should predict the behavior of the reactor in a reliable manner because the code will be used for all analyses.

- In response to configuration control and quality assurance aspects of the computer codes and qualification of Holtec staff on these codes, SMR (Holtec) stated that it is planning to include the codes consistent with its quality assurance program noting

[4] U.S. NRC, Regulatory Guide 1.203, "Transient and Accident Analysis Methods," dated December 2005, ML053500170.

[5] U.S. NRC, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000, ML003716792.

nuances or differences for each computer code, and qualifying individuals on the use of the codes.

- The NRC staff noted the importance of documenting the reasons for the selection of a computer code for an application because each computer code has different limitations and assumptions. Understanding of these limitations and assumptions will support an evaluation of the code's applicability to the design and identification of gaps in the use of the code.

- In reference to the RELAP-3D validation plan on Slide 9, the NRC staff noted the importance to conduct a quality check of any legacy data used including adequate information and documentation of the data prior to its use. The applicability of the data, including a scaling analysis, should be performed prior to using the data for code validation.

- With respect to the completion of Steps 1 through 4 of Element 1 of RG 1.203, the NRC staff requested information on plans to discuss the phenomenon identification and ranking table including the scope of the testing program and scenarios. SMR (Holtec) responded that it plans to request a future discussion on topic.

- In discussing the SMR-160 integral effects test (IET) facility on Slide 10, the NRC noted that scaling distortions are functions of the scaling ratio. The effects of these distortions should be evaluated to show the applicability of data for code validation. NRC staff also questioned the adequacy of IET data from only one facility at one fixed scale ratio. In order to address the scale dependency of data (specifically for high-ranked phenomena) IET data at different scales are desirable (see Section 1.2.3 of RG 1.203).

  Furthermore, the NRC staff noted the importance to consider the containment layout in the test facilities because of the smaller SMR-160 containment compared with existing NPPs. SMR (Holtec) stated that it plans to have a separate effects test (SET) facility for containment such that two different scales of containment will be tested in the IET and SET facilities.

- In response to the NRC staff's question, SMR (Holtec) noted that the startup pump is off at power conditions. SMR (Holtec) noted that the small startup pump and the jet pump effect is minimal and would not result in a transient.

- The NRC staff requested information on the potential for the shutdown pump to create a potential disturbance to the system and SMR (Holtec) responded that this scenario will be analyzed in the future. The NRC staff noted that the pump could cause a sudden change to the flow regime and change reactivity in the core.

- The NRC staff requested information related to 10 CFR 50.62 and whether SMR (Holtec) would be demonstrating compliance with this regulation. SMR (Holtec) responded that it is considering an exemption to parts of the regulation and will request a future discussion with the NRC staff on the topic.[6]

[6] Title 10 to the *Code of Federal Regulations* (10 CFR), 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants."

- SMR (Holtec) confirmed that it is following the RG 1.203 methodology for addressing anticipated transients without scram events.

- In discussing single failures on Slide 15, consistent with SECY-94-084 the NRC staff requested additional information on whether SMR (Holtec) planned to conduct reliability analyses or leverage existing data to exclude consideration of single failures to certain check valves.[7] SMR (Holtec) responded that it plans to request a future discussion on this topic.

- With respect to loss-of-coolant accident (LOCA) methods on Slide 16, the NRC staff noted that the importance of following EMDAP to develop applicable LOCA methodology as several conservatisms in 10 CFR Part 50, Appendix K are probably not applicable to the SMR-160 design.

- There were no questions from members of the public at the end of the open session presentation.

The open session ended at 2:25 PM.

The following provides a non-proprietary summary of the discussion during the closed session of the meeting:

- With respect to the testing programs overview, the NRC staff encouraged SMR (Holtec) to consider the review guidance in the Appendix A to DNRL-2022-01 regarding transient and accident analyses considering the requirements in 10 CFR 50.34(a)(4) which reference 10 CFR 50.46, and 10 CFR 50.35 for the reasonable assurance staff finding that the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.[8, 9, 10, 11]

- The NRC staff noted the following during the discussions:

    - Construction permits issued in the past were supported by prototype reactors to test the concept including 10 CFR Part 50, Appendix K.

    - With respect to long-term cooling following a LOCA and the decoupling of the reactor coolant system (RCS) and containment analysis, the NRC staff noted that decoupling the analysis of these two systems can present challenges in defining the conservative direction. The NRC staff noted that the SMR-160 containment is an integral part of long-term core cooling. If the containment and RCS are tightly coupled during the long-term phase of LOCA transient, determining the conservative direction for many phenomena can be difficult.

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[7] U.S. NRC, SECY-94-084, "Policy and Technical Issues Associated with The Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," March 28, 1994, ML003708068.

[8] U.S. NRC, Interim Staff Guidance (ISG), DNRL-ISG-2022-01, "Safety Review of Light-Water Power-Reactor Construction Permit Applications," November 14, 2022, ML22189A099.

[9] 10 CFR 50.34, "Contents of applications; technical information."

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

[11] 10 CFR 50.35, "Issuance of construction permits."

- With respect to replacing Appendix K correlations with the ones from IET, NRC staff noted that the approval is based on the information submitted for specific scenarios. The NRC staff added that the specific language in the regulation may already allow flexibility to use design-specific correlations without the need for an exemption. However, the applicant needs to verify it on a case-by-case basis.

- With respect to using collapsed level and fuel cladding temperature to meet LOCA post trip acceptance criteria, NRC staff noted that this may be appropriate.

- Based on previous applications, scalability was a concern for IET and SET facilities.

- While probabilistic safety assessments may inform the design to address transients, the NRC staff would need to understand the actual plant response to a transient.

- Having clear conservatisms will simplify the justification for margins.

- At the end of the closed session, the NRC staff commented that it is open to future interactions as the transient and accident analyses methods mature and to discuss the applicability and gaps covered by the methodologies.

The meeting was adjourned at 3:25 PM.

After the meeting, the NRC staff noted the following:

- On Slide 8 regarding the applicability of pressurized-water reactor (PWR) critical heat flux (CHF) methods to the SMR-160 design, the NRC staff noted that the SMR-160 fuel geometry may be similar to PWR fuel geometry. However, SMR-160 is a natural circulation reactor. Consequently, the thermal-hydraulic conditions and CHF mechanism in SMR-160 are expected to be different as compared to conventional PWRs. The SMR-160 CHF methodology would be expected to address these differences.

- With respect to the timeline described on Slide 11 of the Proprietary Slide set, the NRC staff noted that the timeline does not appear to support the proposed schedule for the construction permit application consistent with the guidance in Appendix A of DNRL-ISG-2022-01.[3, 8]