SMR, LLC (Holtec) SMR-160
Staff Observations Regarding October 19, 2022, Public Meeting
Materials for a Potential Exemption from Loss-of-Coolant Accident
Analyses

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the October 19, 2022, public meeting materials regarding a potential exemption from the loss-of-coolant accident (LOCA) analyses for the SMR-160 design.[1] In reviewing this information, the NRC staff note the following observations. These observations are provided to the applicant before the meeting to facilitate the discussion during the meeting.

General Comment

The NRC staff understands that the purpose of the document, "… is to capture and document the potential items that will be used to assist in justifying an exemption to the NRC regulations governing postulated Loss of Coolant Accidents (LOCAs) for the SMR-160 Reactor Coolant System (RCS). This report lists these items in 'Table of Contents' format which will be used to develop a report containing the justification for the exemption request."

1. What is the form/format of the future report? Does the applicant plan to submit a White Paper or a Topical Report with conditions and limitations to support the exemption justification in the construction permit application?

2. With respect to the information provided in "Table of Contents" format, the NRC staff cautions the applicant that the feedback provided does not preclude the NRC staff's request for additional information during the evaluation of an application for specific technical details.

NRC Staff Observations on Sections in the Document

2.1 SMR-160 RCS and Subject Locations
Description of SMR-160 Design (including RCS and ECCS Systems)

The NRC staff presumes that "subject locations" are the "LOCA locations" or "pipe break locations." However, the use of "Subject locations" is confusing when the terminology is used in various 2.3.X Sections of the report. For example, in Section 2.3.3, Material Design, the subject locations could mean the PIF locations at the reactor vessel or steam generator or the SG riser locations. The material design information that are required for the PIF are different from the material design information that are required for the SG riser. Each of the 2.3.X sections needs to specifically identify the exact location (component) to which the specifications are applicable.

2.3.1 Systems Design
Discussion of Applicable ASME Codes for the SMR-160 Design (including specifics for the Subject Locations)—

[1] ML22276A068

2.3.2 Mechanical Design

Descriptions and Drawings for the Subject Locations
- Include material labels for the RCS and Subject Locations

—clarify what is meant by “material labels”

Discussion of the Forging Process for the RCS vessels and Subject Locations

The comparison should include the specific PIF welds to the reactor and SG, and not just the forging.

Discussion of System Structural Layout

Clarify that lateral supports for the SG and RV will also be provided to minimize the lateral movement of the PIF during a seismic event and during a LOCA event.

Discussion of Nondestructive Examination for Fabrication, Preservice and Inservice

The discussion and commitment should include objective evidence that these welds can achieve 100 percent volumetric examination (ultrasonic inspection), for example, the PIF-to-SG weld which is a corner butt weld with reinforcing fillet weld which is similar to branch connections (weld-o-lets) which have experienced limited examination coverage in operating reactors.

2.3.3 Material Design

Discussion of the Fabrication of Materials (including Welding Methods, Weld Residual Stresses and Analysis, Weld Controls/Qualifications, Post Weld Heat Treatment)

The recent operating experience of existing nuclear plant has shown that the post-weld heat treatment (PWHT) of welds in certain steam generators and pressurizers deviated from the requirements of the ASME Code, Section III. The temperature of the PWHT applied to the steam generator and pressurizer welds was lower than the temperature required by the ASME Code, Section III. Include a discussion on how Holtec will ensure that the PWHT temperature will follow the requirements of the ASME Code, Section III.

The discussion for weld residual stresses should include the PIF to SG and SG riser to tubesheet welds and how highly restrained areas (such as field welds, or thick to thinner members) may affect residual stresses.

Discuss Potential Fatigue Issues with the Subject Locations

The operating experience of existing nuclear plants has shown that thermal stratification in pipes cause flaws. Discuss whether thermal stratification will occur in the inner and outer duct (PIF) design in the SMR-160 based on operating experience of existing nuclear plants.

Discuss Potential Thermal (Aging) and Hydrogen Embrittlement of the Subject Locations. Note: Thermal may not apply if not cast materials.

The discussion for hydrogen embrittlement should be clarified to commit to specific industry guidance to be followed to prevent hydrogen embrittlement.