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U.S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

Subject: SMR, LLC Submittal of Additional Preapplication Meeting Materials for September 13, 2022 (Project No. 99902049)

SMR, LLC is pleased to submit additional supporting documentation for a preapplication meeting regarding the control rod drive system (CRDS) to support the construction permit application for the SMR-160 design that was previously held on September 13, 2022.

If you have any questions or require any additional information, please contact Justin Hawkins, SMR-160 Director of Licensing, at j.hawkins@holtec.com, (O) 856-797-0900 x3452, or (C) 609-941-5765.

Respectfully,

Justin Hawkins

Justin Hawkins
Director of Licensing, SMR, LLC

Enclosures:

1. HI-2188193-R1, SMR-160 Control Rod Assembly Design Criteria Document

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Revision Log

Revision	Description of Changes
0	Initial issue.
1	Content of this document is changed from a control blade design to a control rod design. Changes are marked with revision bars except for the report footer/header.

Table of Contents

1.0 INTRODUCTION

1.1 Purpose

This document describes the design requirements and regulatory basis, along with relevant acceptance criteria, for the control rod assembly component of the SMR-160.

1.2 Scope

The control rod assembly (CRA) is a subset of the control rod drive system (CRDS). The CRA connects to the control rod drive (CRD) via a coupling interface(s) to the CRDM. This document specifies the requirements of the CRA to be used for design and analysis. Details of the other subsystems are provided in Reference [2].

1.3 Definitions and Acronyms

A master list of terms and acronyms can be found in SMR-160 Project Systems List, Acronyms and Glossary of Terms [1].

2.0 DESIGN OVERVIEW

The control rod assembly is a component of the SMR-160 reactor. The primary functions of the control rod assembly are to compensate for reactivity changes associated with normal operations and to be capable of bringing the reactor to and maintaining it in a shutdown condition.

The control rod assembly consists of a rod cluster with absorber material and a top connector piece that can be coupled to a drive shaft to the CRDM in order to handle, remove, and insert the control rod assembly during operation. The rod cluster is comprised of steel tubes (e.g., stainless or Inconel) filled with neutron absorbing material (e.g. B_4C, AIC, or hybrid). A conceptual depiction of the control rod assembly is provided in **Error! Reference source not found.**.

3.0 SAFETY FUNCTIONS

This section contains the safety functions of the control rod assembly. The classification of plant states are broken up broadly into: normal operation, anticipated operational occurrences (AOOs), and postulated accidents (PAs). The primary functions are as follows:

1) To compensate and regulate for reactivity during normal operation
2) To be able to start-up and shutdown the reactor during normal operation
3) To be able to shutdown the reactor and maintain it in a subcritical state during AOOs and PAs.

4.0 REGULATORY BASIS

The SMR-160 is a pressurized light water reactor that is similar in many respects to existing reactors. United States Nuclear Regulatory Commission (NRC) source documents were used as guidance.

Under the provisions of 10 CFR 50.34 [3], an application for a construction permit must include the design criteria for a proposed facility. The General Design Criteria (GDC) under 10 CFR 50 Appendix A [4] establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission.

Specified codes and standards are used to establish requirements for construction of the applicable portions of the CRA. Individual codes and standards each provide a set of applicable limits that the design must meet in order to ensure that the applicable component can carry out its designated safety function.

The following regulations are relevant to the control rod assembly:

- 10 CFR 50 Appendix A, GDC 10

  The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDL) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

- 10 CFR 50 Appendix A, GDC 26

  Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

- 10 CFR 50 Appendix A, GDC 27 and as it relates to 10 CFR 50.46 [5]

  The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

- 10 CFR 50 Appendix A, GDC 28

  The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

5.0 LICENSING ACCEPTANCE CRITERIA

To ensure compliance with regulatory requirements in the GDCs, the guidance provided in NUREG-0800 “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants” [6] is followed. The sections most pertinent to the design and performance of the CRA are:

Section 4.2 “Fuel System Design,” and

Section 4.3 “Nuclear Design”

Beyond the scope of this document is Section 3.9.4 “Control Rod Drive System,” Section 4.5.1 “Control Rod Drive Structural Materials,” and Section 4.6 “Functional Design of Control Rod Drive System.”

SRP Section 4.2 [5] identifies that the fuel system, where CRA is included, safety analysis must provide assurance of the following:

1. Fuel system damage as a result of normal operation and AOOs will not be severe enough to prevent the insertion of control rod assemblies
2. The core will always be maintained in a coolable geometry

GDC 10 establishes SAFDLs for all known damage mechanisms that should not be exceeded during any condition of normal operation, including the effects of AOOs. Therefore, SAFDLs are established to ensure that the fuel is not damaged. Within this context, “not damaged” means that the fuel rods do not fail, fuel system dimensions remain within operational tolerances, and functional capabilities are not reduced below those assumed in the safety analysis. “Coolability,” in general means that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat even after a severe accident. The general requirements to maintain control rod assembly insertability and core coolability appear GDC 27 and 10 CFR 50.46 [5].

SRP Section 4.3 [5] identifies the nuclear design of the control systems and reactor core is carried out to aid in confirming that fuel design limits will not be exceeded during normal operation or anticipated operational transients and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary or impair the capability to cool the core and to assure conformance with the requirements of GDC 10, 26, 27, and 28.

Under the envelope of SRP [6], specific acceptance criteria acceptable to meet the relevant requirements of the NRC’s regulations[1] (identified in Section 4 above) are stated in Sections 5.1 through 5.6.

5.1 Functional Requirements

5.1.1 The system of all control rod assemblies shall have sufficient worth to maintain the reactor in a subcritical state during cold conditions at all times, assuming the highest reactivity worth control rod assembly is stuck fully withdrawn from the reactor.

5.1.2 The control rod assembly shall be designed to ensure that it can be inserted during and remain inserted after AOOs and PAs. This include, at minimum, control rod or fuel deformation and vibration due safe shutdown earthquake.

[1] The SRP is not a substitute for the NRC’s regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

5.1.3 The degree of damage to the control rod assembly from AOOs and PAs must not result in reduced reactivity worth of the control rod assembly.

5.1.4 Subsequent to a reactor scram (i.e., rapid insertion of control rod assemblies to shut down reactor), the control rod assembly design and interface with other reactor internals shall be capable of slowing down the control rod assembly to acceptable velocities, ensuring no damage to control rod assembly or other reactor components, before reaching the guide tube lower-end fitting.

5.1.5 The design of the control rod assembly shall meet the applicable ASME code limits during all service conditions including normal and abnormal operations.

5.1.6 The control rod assembly design shall preclude sharp edges, protrusions and other features that adversely impact the ability to insert or retrieve interfacing components.

5.1.7 The control rod assembly design shall minimize areas that could break off and produce wear products.

5.1.8 The control rod assembly bow shall not be so severe as to prevent insertion when required. The effects shall include, at minimum, manufacturing tolerances, swelling, and irradiation growth.

5.1.9 The control rod assembly shall function under repeated cycles of scram events.

5.2 Material Requirements

5.2.1 The material of the control rod assembly appurtenances, attached to or integral with the control rod assembly, shall be compatible with the reactor coolant chemistry requirements.

5.2.2 The maximum tube internal pressure shall not result in tube burst under irradiation of the absorber material.

5.2.3 The maximum metal thinning due to oxide growth of the control rod assembly shall not result in stress or strain limits being exceeded.

5.2.4 The maximum allowable hydrogen content of the control rod assembly shall not cause in ductility being reduced to the extent that would result in stress or strain limits being exceeded.

5.2.5 The metallurgical structure and composition shall be sufficiently uniform to preclude differential irradiation growth or creep.

5.3 Environmental Requirements

5.3.1 The materials chosen for the control rod assembly design shall be shown to be compatible with the environment conditions of the reactor coolant system.

5.4 Thermal-Mechanical Requirements

5.4.1 The control rod assembly shall meet the applicable design and service load combinations, including the most severe combination of applicable loads.

5.4.2 The control rod assembly structural components stresses shall not exceed the stress limits for AOOs. For PAs, alternative stress limits may be applied in specific analysis as long as the control rod assembly can still perform its safety functions. Stress limits are obtained by methods similar to those given in the ASME Boiler and Pressure Vessel Code, Section III [7].

5.4.3 The control rod assembly structural components strain shall not exceed the strain limit for AOOs. For PAs, strain limits may be applied in specific analysis as long as the control rod assembly can still perform its safety functions. Strain limits are obtained by methods similar to those given in the ASME Boiler and Pressure Vessel Code, Section III [8].

5.4.4 The control rod assembly structural components strain fatigue shall not exceed a cumulative usage factor limit for AOOs. The number of scrams and seismic cycles shall be included over the life of the CRA. For PAs, alternative limits may be applied in specific analysis as long as the control rod assembly can still perform its safety functions.

5.5 Interface Requirements

5.5.1 The control rod assembly shall connect to the CRDM drive rod by means of a mechanical coupling and shall allow for disconnection by either tool or other means.

5.5.2 Interfaces between the control rod assembly the following components shall be defined:

1) Guide tubes
2) Upper core plate
3) Upper internals
4) Spent fuel storage rack

5.6 Operations and Maintenance Requirements

5.6.1 The control rod assembly shall be designed to operate for a prescribed irradiation time.

5.6.2 The control rod assembly shall be designed to operate until a prescribed cold reactivity worth or stress/strain limits are violated.

5.6.3 The control rod assembly shall be designed to be replaceable.

5.6.4 The control rod assembly shall be designed to be remotely handled.

6.0 REFERENCES

[1] Holtec International Report HI-2146109, SMR-160 Project Systems List, Acronyms, and Glossary of Terms, Revision 4.

[2] System Design Description for Control Rod Drive System, HI-2200941, Revision 0, Holtec International.

[3] 10 CRR 50.34, “Contents of Applications; Technical Information”.

[4] 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants".

[5] 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors".

[6] NUREG-0800, " Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” U.S. Nuclear Regulatory Commission, 2007.

[7] ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components, 2017 Edition.

[8] Control Rod Assembly General Arrangement, DWG-11900, Revision 0, Holtec International.