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6 Safety-Related Structure, System, and Component Criteria and Capabilities

6.1 Design Requirements for Safety-Related Structures, Systems, and Components

Preliminary safety analysis report (PSAR) section 6.1 describes the determination of structure, system, and component (SSC)-specific design criteria and provides details of the analyses and justifications for the development of safety-related design criteria (SRDC). PSAR section 6.1.1 discusses the design basis hazard levels (DBHLs), PSAR section 6.1.2 summarizes the SRDC, and PSAR section 6.1.3 summarizes the DBHL-related requirements for non-safety-related SSCs.

The applicable regulatory requirements for the design requirements for safety-related SSCs are as follows:

- 10 CFR 50.34(a)(3)(ii)
- 10 CFR 50.34(a)(4)

The applicable guidance for the review of the design requirements for safety-related SSCs are as follows:

- RG 1.76, Rev. 1, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants." (ML070360253)
- RG 1.115, Rev. 2, "Protection Against Turbine Missiles." (ML101650675)
- RG 1.189, Rev. 5, "Fire Protection for Nuclear Power Plants." (ML23214A287)
- RG 1.221, Rev. 0, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants." (ML110940300)
- RG 1.233, Rev. 0, "Guidance for a Technology Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors" (ML20091L698), which endorses, with clarifications, NEI 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Revision 1 (ML19241A336)
- RG 1.247, Rev. 0, "Acceptability of Probabilistic Risk Assessment Results for Non-Light-Water Reactor Risk-Informed Activities," For Trial Use (ML21235A008)
- RG 1.253, Rev. 0, "Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors" (ML23269A222) which endorses, with certain clarifications and additions, the guidance in NEI 21-07 regarding the content of applications for designs that follow the process described in NEI 18-04 and provides guidance regarding implementation of the NEI 18-04 methodologies.

- DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications – Roadmap." (ML23277A139)

6.1.1 Technical Evaluation

Design Basis Hazard Levels

The NRC staff reviewed the description of DBHLs in PSAR section 6.1.1 and table 6.1-1 using applicable guidance in RG 1.233 and RG 1.253, which endorse, respectively, NEI 18-04 and, with certain clarifications and additions, NEI 21-07. Per NEI 21-07, the construction permit (CP) application content for section 6.1.1, should be as complete as possible based on the site characterization required in 10 CFR 50.34(a). NEI 21-07 further states that DBHLs include both external and internal hazards, such as seismic events, wind (including tornadoes and wind-generated missiles), external flooding, hazards from external facilities, internal fires, internal floods, high-energy line breaks, and internally generated missiles.

The NRC staff review determined that LME selected a set of DBHLs that is consistent with NEI 21-07. For each hazard, the NRC staff evaluated the information provided in the PSAR to ensure it was reasonable to establish the design bases for protection against hazards for applicable SSCs consistent with PDCs 2, 3, and 4. For the CP application, LME characterized the external DBHLs using the results from PSAR Chapter 2, "Site Characteristics." For internal hazards that rely on final design information (e.g., internal fire, internal flood, HELB), the NRC staff assessed the applicant's plans to ensure safety significant SSCs will have appropriate protection against these hazards at the operating license (OL) application stage. PSAR table 6.1-1 provides a summary of the DBHLs. The NRC staff's evaluation of each hazard is described below.

Seismic Hazard

The NRC staff reviewed PSAR section 2.5.2, which addresses the seismic hazard at the LMGS site, and PSAR sections 3.7, 3.8, 6.4, and 7.3, which describe the associated seismic analysis methodology and design. The horizontal ground motion response spectrum (GMRS) for the LMGS site has a peak ground acceleration (PGA) lower than 0.1g and therefore the safe shutdown earthquake (SSE) horizontal design response spectrum was increased to achieve a 0.1g PGA in accordance with the minimum PGA required in Appendix S to Part 50. The NRC staff's review of the LMGS seismic hazard and evaluation of the potential impact of a seismic event on SSCs are provided in sections 2.5.2, 3.7, 3.8, 6.4, and 7.3 of this safety evaluation (SE). In those sections, the NRC staff determined that LME has provided the necessary information to address the design-basis seismic hazard and to support its evaluation of how the design addresses the hazard.

Furthermore, as described in PSAR section 3.7.3.3, LME will perform a seismic probabilistic risk assessment (SPRA) in support of a future OL application to confirm the overall seismic performance and/or update seismic design criteria as necessary. Although not explicitly considered in the DBHLs, sufficient seismic design margin would need to be demonstrated at the OL application stage to meet the licensing modernization project (LMP) risk target for beyond-design-basis events (BDBEs). The applicant's preliminary approach is consistent with the discussion in RG 1.253, Appendix A.3.7, which states that certain hazards (e.g., external hazards such as seismic hazards) may be dispositioned at the CP stage by crediting DBHLs in

lieu of explicitly modeling them in the PRA and evaluating their consequences. Consistent with RG 1.253, the NRC staff will confirm the results are acceptable against the frequency-consequence (F-C) target when the full scope PRA has been completed at the OL application stage.

Based on the NRC staff's review of PSAR sections 2.5.2, 3.7, 3.8, 6.4, and 7.3 and its findings documented in sections 2.5.2, 3.7, 3.8, 6.4, and 7.3 of this SE, respectively, the NRC staff determined that the DBHL established for the seismic hazard is consistent with the design basis seismic characteristics and is therefore consistent with PDC 2.

Wind Hazard (Tornado and Hurricane Loads and Generated Missiles)

The NRC staff reviewed PSAR section 2.3, table 2.0-1, and section 3.7, which provide the site-specific parameters and applicant's assessment of wind hazards. The wind hazard includes a basic wind speed, a maximum wind speed bounded by the hurricane hazard, a pressure drop and rate of pressure drop bounded by the tornado hazard, and tornado and hurricane generated missiles generated from the missile spectra in RGs 1.76, Rev. 1 and RG 1.221. The NRC staff determined, in sections 2.6 and 3.7 of this SE, that the site-specific wind hazard parameters and associated applicant's assessment are consistent with RG 1.76, Rev. 1 and RG 1.221. Further, the staff's review of PSAR section 6.1.1 and table 6.1-1 determined that the DBHL established for wind hazard is consistent with the design basis wind parameters. On this basis and based on the staff's determinations documented in sections 2.6 and 3.7 of this SE, the staff finds the wind hazard DBHL meets the requirements of PDC 2.

External Flood

The NRC staff reviewed PSAR section 2.4.5.6, which provides flooding information for the vicinity of the LMGS site. The DBHL flood level described in PSAR table 6.1-1 and PSAR table 2.0-1 is bounded by the probable maximum storm surge and includes a still water level and wave run-up. PSAR section 6.1.1.4 describes LME's approach to ensure adequate protection against ingress of water into the buildings due to the flooding DBHL. The approach includes a combination of strategies such as elevation protection (locating systems and components above flood height), waterproof protection, and functional protection (submergence qualification).

The NRC staff determined that the DBHL established for external flooding is consistent with the design basis external flood parameters (as documented in section 2.4 of this SE) and is therefore consistent with PDC 2. Further, the NRC staff determined that the strategies to protect safety-related (SR) SSCs against ingress of water due to external flooding are reasonable because they account for the flooding DBHL flood height, the location of SSCs relative to the flood height, and qualification of systems and components to withstand the effects of the flooding DBHL, as needed, to ensure that they remain capable of performing their required safety functions (RSFs).

Internal Fire

The NRC staff reviewed PSAR section 6.1.1.5, which states that, at the preliminary design phase, fire related PDCs were identified to establish proper design requirements for SSCs. The PSAR states that internal fire hazards will be fully evaluated at the OL application stage when detailed SSC design and configurations are available. The fire scenario selection will include the

identification of a fire ignition source (or set of fire ignition sources), secondary combustibles and fire spread paths, fire damage targets, detection and suppression systems, and features to be credited. PSAR section 8.6 provides additional discussion on fire protection and fire analysis. PSAR sections 6.1.1.5 and 8.6 describe that the internal fire hazards and planned features will be developed in conformance with applicable guidance, including RG 1.189, NFPA 804, and the non-LWR PRA standard, which is endorsed by RG 1.247, with any deviations described and justified.

The NRC staff determined that the internal fire hazard at LMGS is reasonably characterized for the CP stage, and that planned features discussed in the PSAR are appropriate to ensure that SR SSCs will maintain their safety-significant functions under postulated internal fire conditions, consistent with PDC 3. The NRC staff will review LME's evaluation of the fire protection features at the OL application stage based on the final design information.

Internal Flood

The NRC staff reviewed PSAR section 6.1.1.6, which states that an internal flood event will be expressed in terms of the specific flood source and its local impact. PSAR section 6.1.1.6 identifies potential design features to protect safety-significant SSCs from postulated flooding, including isolation valves, physical barriers, and drains. The specific flood protection features will be developed as part of the OL application. Additionally, at this preliminary stage, LME identified several SR SSCs as not being subject to local flood sources. At the OL application stage, LME will confirm that SSCs will either not be subject to any local flood source or will be designed to not fail or challenge any RSFs.

The NRC staff determined that the internal flooding hazard at LMGS is reasonably characterized for the CP stage, and that planned design measures discussed in the PSAR are appropriate to ensure SR SSCs will maintain their safety-significant functions under postulated internal flood conditions, in accordance with PDC 4. The NRC staff will review LME's evaluation of the necessary flood protection design features at the OL application stage based on the final design information.

High-Energy Line Breaks (HELBs)

The NRC staff reviewed PSAR section 6.1.1.7, which states that HELB hazards are characterized as a break of a pipe or piping system component containing a fluid maximum operating temperature exceeding 200°F or a maximum operating pressure exceeding 275 psig. Further, LME identified several SR SSCs that are either shielded from or are not subject to HELB sources. PSAR section 6.1.1.7 states SR SSCs will be designed such that they will be able to withstand or be protected from the impact of HELB missiles without any adverse impact on their capability to perform required SR functions. The NRC staff will review analyses based on the final design as part of a future OL application review to confirm that SR SSCs will be capable of performing their functions. Design-basis pipe ruptures are addressed by one or more of the following methods: physical separation, missile barriers, or demonstrating that HELB missile impact does not adversely impact the ability of SR SSCs to perform their RSFs.

The NRC staff determined that the HELB hazard at LMGS is reasonably characterized for the CP stage, and that planned design measures discussed in the PSAR are appropriate to ensure

SR SSCs will maintain their safety-significant functions under postulated HELB conditions, consistent with PDC 4. The staff will review LME's evaluation of HELBs and their impacts at the OL application stage based on the final design information and piping layout.

Turbine and Internally Generated Missiles

The NRC staff reviewed PSAR section 6.1.1.3, in which LME identified turbine missiles, other missiles from rotating equipment, and secondary missiles outside the scope of the PRA for the CP application. PSAR table 1.4-2 indicates that LME's approach is consistent with RG 1.115 and will consider location and orientation of missile sources, missile generation frequency and frequency of damage to SSCs, and the use of missile barriers. The final assessment of turbine and internally generated missiles will be addressed at the OL application stage via an internal hazards PRA. The NRC staff review determined that LME's approach for the preliminary design is appropriate to ensure SR SSCs will be adequately protected against turbine and other internally generated missiles. The NRC staff will review LME's evaluation of turbine and other internally generated missiles and their impacts at the OL application stage based on the final plant layout and design information.

Hazards From External Facilities

LME addressed hazards from external facilities in PSAR section 2.2. The NRC staff evaluated these hazards in section 2.5 of this SE and concluded that LME has shown that hazards from nearby external facilities and transportation routes are not expected to develop into a design-basis hazard. The staff noted an exception related to general aviation aircraft crashes for which LME committed to performing a final assessment of accidental aircraft hazards at the OL stage. The NRC staff also noted that the preliminary assessment performed by LME is conservative and determined that the review of the final assessment of accidental aircraft impact can be reasonably left for the OL stage.

Based on its review of the preliminary assessments and design approaches discussed above, the NRC staff finds that the DBHLs above are reasonable. This finding is based on LME's approach to defining the DBHLs for the preliminary design being consistent with the guidance in RG 1.233, RG 1.253, and DANU-ISG-2022-01. As more detailed design and final information becomes available, the applicant will further assess the internal flooding, internal fire, high-energy line break hazards, hazards related to turbine and other internally generated missiles, and hazards from external facilities, to confirm and update their associated DBHLs as necessary and applicable. The staff will review LME's final evaluation of the internal hazards at the OL application stage, based on the final design information and plant layout.

Summary of SRDCs

The NRC staff reviewed PSAR section 6.1.2, which summarizes the SRDCs. The SRDCs are used to map RSFs and required functional design criteria (RFDCs) to specific SR SSCs. LME provided a list of SRDCs in PSAR table 6.1-2. Specifically, this table lists the SR SSCs, the RSFs that the SR SSCs fulfill, and the RFDCs and SRDCs corresponding to each SR SSC. The NRC staff review determined that the information provided in PSAR table 6.1-2 is consistent with the guidance in section 6.1.2 of NEI 21-07 and therefore is acceptable. Further, the NRC staff's review confirmed that the SRDCs listed in the table are addressed in the respective

system subsections in PSAR chapter 6. The NRC staff's evaluation of the SRDCs is addressed in the respective SR system discussions in section 6.4 of this SE.

Summary of DBHL-Related Requirements for Non-Safety-Related SSCs

The NRC staff evaluated PSAR section 6.1.3, which summarizes the DBHL-related requirements for NSRST and NST SSCs. These requirements are applied to non-safety-related SSCs to prevent adverse impacts on SR SSCs to perform their RSFs and do not affect the safety classification of the SSCs (which are based on the functions that the SSCs perform within the safety analysis). LME's approach for preventing potential adverse impacts of non-safety-related SSCs on the RSFs of SR SSCs includes physical separation, adjacent protection barriers, or designing the SR SSCs to withstand physical or functional impact of the failure of the NSRST or NST SSCs. Also, PSAR section 3.7 indicates that NSRST and NST SSCs are classified as Seismic Category IIB and are designed to prevent adverse impacts on SR SSCs. Additionally, the design criteria for NSRST structures include extreme tornado/hurricane design loads and missile impacts in accordance with the concrete and steel nuclear codes and standards and associated regulatory guides. The NRC staff review found LME's approach for preventing potential adverse impacts on SR SSCs to be consistent with the guidance in DANU-ISG-2022-01.

A preliminary list of non-safety-related SSCs with DBHL-related requirements is provided in PSAR table 6.1-3. Furthermore, PSAR section 6.1.3.8 describes LME's process for identifying additional NSRST and NST SSCs to which DBHL-related requirements must be applied to prevent adverse impacts on the ability of SR SSCs to perform their RSFs. Per this process, hazards sources are mapped to SR SSC targets by considering physical layouts, PSF functional dependencies, prevention and mitigation capabilities, the availability of supporting equipment, plant operating states, and plant work activities, among others. Primary sources and primary-sources-dependent secondary sources are also considered. The process also includes walkdown verifications. The NRC staff review finds this process to be systematic and thorough. Further, the identification and evaluation of NSRST and NST SSCs that need DBHL requirements is consistent with the guidance in RG 1.233 and RG 1.253.

Per PSAR section 3.7.3.1, a subset of NSRST SSCs are categorized as Seismic Category IIA (SC-IIA) based their seismic risk significance consistent with the NEI 18-04 risk metrics. These NSRST SSCs need to perform their function and meet the NEI 18-04 risk metrics during the seismic DBHL. The staff's evaluation of the seismic design response spectra for these SSCs, which are based on the seismic DBHL/SSE for the LMGS site, is addressed in section 3.7 of this SE. Further, these NSRST SSCs are also protected from adverse interactions from other NSRST or NST SSCs. The process for mapping hazard sources to these NSRST SSCs is consistent with that used for SR SSCs. The NRC staff finds the hazards/sources mapping process used for this NSRST SSCs to be acceptable because it is consistent with that used for SR SSCs.

The NRC staff reviewed the PSAR section 6.1.3, which summarizes the DBHLs considered in the determination of DBHL-related requirements for NSRST and NST SSCs. The NRC staff review finds the scope of DBHLs described in PSAR section 6.1.3 to be consistent with those identified in PSAR section 6.1.1. Moreover, PSAR section 6.1.3 describes that probabilistic assessments will be performed as part of a future OL application to confirm or update the

external DBHL-related requirements and develop the detailed internal DBHL-related requirements. The NRC staff review finds LME's approach to use probabilistic assessments in the final selection of DBHLs to be consistent with RG 1.253, which endorses, with certain clarifications and additions, the alternative approaches for the selection of DBHLs described in NEI 21-07, including deterministic and probabilistic approaches. The NRC staff will review the final DBHL-related requirements for NSRST and NST SSCs at the OL application stage.

The NRC staff review also identified that certain NSRST and NST equipment and instrumentation that may fail during the flooding DBHL require the development of strategies and procedures to cope with their unavailability during the flooding DBHL for the LMGS site. These NSRST and NST SSCs include equipment and instrumentation located in the controls and electrical building (CEB). As described in PSAR section 6.1.3.4, LME mitigates the effects of the potential unavailability of these supporting equipment and instrumentation via implementation of its emergency plan and activation of the LMGS emergency response organization. Per LME's emergency plan, the plant is to be placed in a safe shutdown condition either from the main control room (MCR) or from the reactor protection system (RPS) cabinets located in the fuel handling auxiliary building (FHAB) if the MCR becomes unavailable due to the external flood DBHL event. The staff notes that PSAR section 7.3.1.1.1 describes the design requirements and strategies for protecting the FHAB and the SR SSCs within it from the flooding DBHL. The staff's evaluation of such strategies is addressed in SE section 7.3.1. In the case that the MCR becomes unavailable due to the external flood DBHL event, indications from the RPS cabinets as well as portable monitoring equipment would be used to monitor the status and radiological condition of the plant throughout the flooding event. Further, per PSAR section 6.1.3.4, LME will provide detailed emergency plans, emergency operating procedures, and details associated with the portable monitoring equipment and indications at the OL application stage. The NRC staff will review the emergency plan, operating procedures, and details associated with the portable monitoring equipment and indications at the OL application stage.

PSAR chapter 8, "Plant Programs," addresses, in part, the applicability of plant programs to NSRST SSCs and the NRC staff's evaluation of these programs is in chapter 8 of this SE. The application of DBHL requirements to NST SSCs that are necessary to ensure RSFs are performed requires that certain programs be applied to NST SSCs to ensure the requirements placed on those SSCs are met. Per PSAR section 6.1.3, the programmatic controls for NST SSCs will be determined by the integrated decision-making process panel (IDPP) and may include testing, maintenance, and/or inspection controls. Additionally, per PSAR section 8.1.3, NST SSCs have QA requirements established based on ISO 9001. During the audit, the NRC staff confirmed that LME's process for establishing programmatic controls for NST SSCs includes a condition evaluation to determine if additional programs or parts of programs should be applied to NST SSCs. The NRC staff will evaluate the detailed programmatic requirements applied to NST SSCs needing DBHL-related requirements at the OL stage.

6.1.2 Conclusion

Based on its review, the NRC staff finds that the information on DBHLs, SRDCs, and DBHL-related requirements for NSRST and NST SSCs, described in PSAR Section 6.1, is consistent with the applicable regulatory guidance listed above, and meets the relevant portions of 10 CFR 50.34(a)(3)(ii) and 10 CFR 50.34(a)(4). .

6.2 Reliability and Capability Targets for Safety-Related Structures, Systems, and Components

PSAR section 6.2 describes the applicant's approach to developing reliability and capability targets for safety-related (SR) SSCs and presents the set of targets developed based on the preliminary design.

The applicable regulatory requirements for the reliability and capability targets for SR SSCs are as follows:

- 10 CFR 50.34(a)(4)
- 10 CFR 50.35(a)(1)

The applicable guidance for the review of PSAR section 6.2 is as follows:

- RG 1.233, "Guidance for a Technology Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," Rev. 0 (ML20091L698), which endorses, with clarifications, the guidance in NEI 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Rev. 1 (ML19241A472).

- RG 1.253, Rev. 0, "Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," which endorses, with clarifications and additions, NEI 21-07, "Technology Inclusive Guidance for Non-Light Water Reactor Safety Analysis Report: For Applicants Utilizing NEI 18-04 Methodology," Rev. 1 (ML22060A190).

6.2.3 Technical Evaluation

As described in PSAR section 1.3.1, the applicant is implementing the LMP methodology as outlined in RG 1.233, which endorses, with clarifications, NEI 18-04. NEI 18-04 contains guidance on defining the reliability and capability targets as part of the LMP methodology. As described in PSAR chapter 1, LME is generally following the content and organization guidance provided in NEI 21-07 and endorsed, with clarifications, in RG 1.253. NEI 21-07, Revision 1, section C.6.2, "Reliability and Capability Targets for SR SSCs," contains applicable guidance for content and review of section 6.2 of the PSAR.

The NRC staff evaluated PSAR section 6.2 using the guidance in RG 1.233 and RG 1.253. As described in NEI 18-04, reliability and capability targets are used to establish controls on the frequency and consequence levels of non-DBA licensing basis events (LBEs) and to inform the selection of special treatment requirements for SR and NSRST SSCs. LME provides preliminary plant-level reliability and capability targets for SR SSCs in PSAR table 6.2-1 to meet the plant-level safety objectives. The reliability and capability targets established ensure that the design

supports the inputs and assumptions used by the safety analysis and to support staff findings pursuant to 10 CFR 50.34(a)(4) and 50.35(a)(1).

The preliminary targets provided in the PSAR are limited to those for required safety functions (RSFs) performed by SR SSCs that are explicitly modeled in the probabilistic risk assessment (PRA) used to support the PSAR. As described in section 3.1.1 of this SE, the scope of this PRA is limited to at-power internal events for the reactor, and some SSCs are not modeled within the construction permit (CP) stage PRA. The NRC staff determined that limiting the preliminary reliability and capability targets based on the PRA scope is sufficient for the CP application given the preliminary design stage, the additional PRA development that would occur between the CP and the operating license (OL), and LME's commitment to follow the guidance in the NEI 18-04 methodology, which includes defining reliability and capability targets for all safety-significant SSCs.

PSAR Table 6.2-1, "Reliability and Capability Targets for SR SSCs," of the PSAR defines the preliminary reliability and capability targets for the following RSFs:

- RSF 1.1.1.2: Ensure sufficient inherent reactivity feedback by controlling primary heat transport
- RSF 1.1.2: Maintain long-term subcriticality
- RSF 1.2.1: Maintain core heat removal through passive means
- RSF 1.3.1: Isolate water/steam sources
- RSF 1.4.1.1: Control primary pressure

The scope of RSFs identified in PSAR table 6.2-1 is a subset of the RSFs for the LMGS design discussed in PSAR Section 5.2. The PSAR states that RSFs not included in the table are not represented probabilistically in the PRA. The NRC staff notes that the RSFs not included in PSAR table 6.2-1 are as follows:

- RSF 1.1.1.1: Ensure sufficient inherent reactivity feedback to maintain the core radionuclide release limits to acceptable levels
- RSF 1.4.1.2: Limit stress in metallic core components
- RSF 1.4.2: Maintain reactor support geometry

NEI 18-04 states the purpose of the reliability and capability targets is to define the requirements for the reliability and capability for SSCs modeled in the PRA. Therefore, the NRC staff finds that the scope of the RSFs discussed in PSAR table 6.2-1, which is consistent with the scope of RSFs modeled in the PRA at the CP stage, is consistent with RG 1.233, which endorses, with clarifications, NEI 18-04. As noted above, while some SSCs are not modeled within the current PRA, the PRA will continue to develop from CP to OL and SSCs that play a role in preventing or mitigating LBEs would be incorporated into the PRA prior to the OL application.

The applicant proposed the following reliability targets for each RSF listed in PSAR table 6.2-1:

- Maintain frequency of all LBEs with failure of the RSF below 1E-4 per plant calendar year.
- Maintain LBEs with failure of the RSF within the F-C target.
- Maintain cumulative risk metrics within targets.

The applicant stated that the basis for the functional-level reliability targets is derived from the development of frequency targets for a collection of LBEs that are prevented or mitigated by a particular RSF. However, the reliability targets listed above from PSAR table 6.2-1 align with the overall requirements that the plant must meet, without further definition of the reliability various functions will need to achieve. Therefore, because the reliability targets align with the overall plant requirements, the NRC staff determined that the preliminary reliability targets defined are aligned with plant level targets, as defined in section 6.2 of NEI 21-07.

NEI 21-07 states that reliability and capability targets can be established at different levels: plant level, functional level, or SSC level. Thus, the applicant's approach of establishing plant-level targets at the CP stage with a defined methodology for establishing functional and SSC-level targets at the OL stage is consistent with RG 1.253, which endorses, with certain clarifications and additions, NEI 21-07. The plant-level targets are discussed further in PSAR chapter 3, "Methodologies, Analyses, and Licensing Basis Events," and chapter 4, "Integrated Evaluations," associated with the frequency-consequence target and cumulative risk metrics.

The LME process for developing functional- and system-level reliability targets from the plant-level targets is described in PSAR section 6.2. This process involves associating the functions with the LBEs they impact and then optimizing the set as a whole to determine the frequency target for each function that maximizes margin between the PRA modeled performance and the target performance impacting the conclusions of the safety analysis. The results from exercising this method for setting reliability targets will be evaluated at the OL application stage consistent with NEI 21-07. The applicant also states that the Final Safety Analysis Report (FSAR) submitted as part of a future OL application will describe system-level reliability and capability targets.

The applicant also proposed a capability target for each RSF identified in PSAR table 6.2-1. The PSAR states that the capability target is informed by the success criteria used in the PRA to determine success or failure of a particular RSF at the functional level and aligns with the corresponding reliability target.

The NRC staff reviewed the proposed preliminary reliability and capability targets and determined them to be reasonable and acceptable based on the following:

1. The preliminary reliability and capability targets for the RSFs and associated SR SSCs support the overall plant-level objectives discussed in the PSAR.
2. Defining plant-level and functional-level targets, instead of SSC-level targets, at the CP stage is reasonable because of the preliminary nature of the LMGS design and the

additional development that is expected to occur between the CP and OL application, as described above.

3. The applicant commits to include the system-level reliability and capability targets in the FSAR as part of the OL application.

6.2.4 Conclusion

Based on its review, the NRC staff finds that the information on the reliability and capability targets for SR SSCs, described in PSAR Section 6.2, is consistent with the applicable regulatory guidance and meets the regulatory requirements under 10 CFR 50.34(a)(4). As discussed in section 6.3 of this SE, the reliability and capability targets for the SR SSCs are principal architectural and engineering criteria for the design under 10 CFR 50.35(a)(1).

6.3 Special Treatment Requirements for Safety-Related SSCs

PSAR section 6.3 describes the applicant's approach to selecting special treatment requirements for safety-related SSCs. Special treatment requirements as defined in NEI 18-04 and stated in PSAR section 6.3 are "requirements that provide increased assurance beyond normal industrial practices that structures, systems, and components (SSCs) perform their design-basis functions."

The applicable regulatory requirements for the evaluation of safety-related SSCs are as follows:

- 10 CFR 50.34(a)(4)
- 10 CFR 50.35(a)(1)

The applicable guidance for the review of PSAR section 6.3 is as follows:

- RG 1.233, "Guidance for a Technology Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors" Rev. 0 (ML20091L698), which endorses, with certain clarifications, NEI 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Rev. 1 (ML19241A472).

- DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap," Section 1.1, "SAR Chapters 1-8".

- RG 1.253, Rev. 0, "Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," which endorses, with clarifications and additions, NEI 21-07, "Technology Inclusive Guidance for Non-Light Water Reactor Safety Analysis Report: For Applicants Utilizing NEI 18-04 Methodology," Rev. 1 (ML22060A190).

6.3.1 Technical Evaluation

RG 1.253, which endorses, with certain clarifications and additions, NEI 21-07, Rev. 1, provides applicable guidance in section C.6.3, "Special Treatment Requirements for SR SSCs," for the content and review of PSAR section 6.3.

PSAR section 6.3 states that the systematic process that LME uses for special treatment selection will be presented in the FSAR. The systematic process for selecting special treatments for each safety-related SSC considers the required safety functions (RSFs) and NSRST PRA safety functions (NSRST PSFs) performed by the SSCs and their associated reliability and capability targets. The IDPP reviews and approves the selection of special treatments.

The NRC staff evaluated the preliminary process described in the PSAR and supplemented its understanding through a focused audit engagement. During the focused audit engagement, the NRC staff and LME engaged in a discussion of how this process is implemented by LME. The NRC staff confirmed through this audit engagement that the preliminary process is being used to systematically evaluate the special treatments being applied to each system with SR components. The process also includes an IDPP review of design documents prior to a review meeting, which includes representatives from safety, engineering, operations, maintenance, etc.

As stated in PSAR section 6.3, the special treatments are derived from the reliability and capability targets for the SSCs, which consider each SSC's performance during every applicable LBE. Based on this statement and the associated description, the NRC staff determined that the performance targets and special treatments will be consistent with what is being relied on in the safety analysis. These statements on the alignment with the full scope of LBEs were necessary based on the limited scope used for the PDCs, CDCs, RFDCs, and SRDCs, as discussed in section 5.3 of this SE. As additional criteria established to address inconsistencies between the PDCs and the safety analysis, the reliability and capability targets and special treatments are principal architectural and engineering criteria for the design under 10 CFR 50.35(a)(1).

A preliminary set of special treatments are described in the SSC descriptions within PSAR section 6.4. For SR SSCs, nuclear codes and standards are generally selected for design. The use of nuclear codes and standards is considered a special treatment in the NEI 18-04 methodology. Additionally, application of the programs described in chapter 8 of the PSAR (e.g., quality assurance, reliability integrity management, inservice testing) are also considered special treatments by NEI 18-04. Because the expectation is that nuclear codes and standards and the programs in chapter 8 will be applied to the SR SSCs, technical evaluations in section 6.4 of this SE note where nuclear codes and standards or programmatic special treatments were not applied as expected rather than noting each place where special treatments are applied. Specific findings with respect to the special treatments applied to individual systems are documented in the corresponding evaluations within section 6.4 of this SE.

A final set of special treatments for SR SSCs is expected to be summarized in this section of the FSAR, in addition to being provided in the corresponding SR system subsections of the FSAR.

The NRC staff reviewed PSAR section 6.3 against the guidance in NEI 21-07. The preliminary special treatments described in the PSAR 6.4 subsections are sufficient for the CP stage because they help ensure the systems can perform the RSFs they are being relied on for, consistent with the preliminary state of the design. The format and content presented in PSAR

table 6.3-1 is consistent with Table 6-4 of NEI 21-07 and provides a reasonable approach to summarize the special treatments in the FSAR.

6.3.2 Conclusion

The NRC staff reviewed the approach to assigning special treatment requirements to safety-related SSCs described in PSAR section 6.3 and determined that the approach is acceptable because LME is using a systematic approach that is consistent with the methodology in NEI 18-04, endorsed with clarifications by RG 1.233. This description is sufficient to address the methodology for the preliminary analysis and evaluation of the design and performance of SSCs of the facility as required by 10 CFR 50.34(a)(4). As discussed in the evaluation above, the special treatments for the SR SSCs are principal architectural and engineering criteria for the design under 10 CFR 50.35(a)(1).

The content of the PSAR is sufficient for the CP stage based on the special treatments described as part of the preliminary design and based on the format and content proposed for section 6.3 of the FSAR, as shown in PSAR table 6.3-1, being reasonable and consistent with the guidance in NEI 21-07, as endorsed with clarifications and additions in RG 1.253.

6.4 Descriptions for Safety-Related Structures, Systems, and Components

PSAR section 6.4 provides descriptions for safety-related (SR) SSCs. For systems that include both SR and non-safety-related with special treatment (NSRST) SSCs, the NSRST SSCs are described in PSAR section 6.4 and evaluated in this section of the SE, along with the SR SSCs for that system. Each subsection in PSAR section 6.4 generally includes the following sections:

- Functions and design criteria: lists required safety functions (RSFs) and NSRST PRA safety functions (PSFs), lists associated principal design criteria (PDCs), provides the system specific safety-related design criteria (SRDCs) and complementary design criteria (CDCs), and then describes how the preliminary design performs these functions and meets these criteria.
- Performance and operation: provides additional discussion on general performance and operation aspects for the system.
- Qualification, testing, and programs: describes the planned qualification and any testing to be performed, including first-of-a-kind (FOAK) testing and research and development (R&D), and describes the preliminary special treatments selected, including programmatic special treatments, for the system.

The SE evaluation section number corresponds to the PSAR section being evaluated. Each evaluation section includes discussion on the safety significance and generally describes the impact of the system failing to perform its function, an assessment of compliance against the PDCs, and a description of the safety functions relied on in the safety analysis.

The applicable regulatory requirements for the design of SR and NSRST SSCs evaluated in sections 6.4.1 through 6.4.14 of this SE are as follows:

- 10 CFR 50.34(a)(3)

- 10 CFR 50.34(a)(4)
- 10 CFR 50.34(a)(5)
- 10 CFR 50.35(a)

System specific regulatory requirements will be listed, as applicable, in the subsection containing the system specific evaluation.

The applicable guidance for the design of SR and NSRST SSCs is as follows:

- RG 1.233, "Guidance for a Technology Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," (ML20091L698) which endorses, with certain clarifications, NEI 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Revision 1 (ML19241A472), with clarifications.
- RG 1.253, "Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," (ML23269A222) which endorses, with clarifications and additions, NEI 21-07, "Technology Inclusive Guidance for Non-Light Water Reactors, Safety Analysis Report Content: For Applicants Using the NEI 18-04 Methodology," Revision 1 (ML22060A190).
- DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications-Roadmap."

System specific guidance will be listed, as applicable, in the subsections containing the system specific evaluation.

PDCs applicable to all systems containing SR and NSRST include:

- PDC 1 – Quality standards and records
- PDC 2 – Design bases for protection against natural phenomenon
- PDC 3 – Fire protection
- PDC 4 – Environmental and dynamic effects design basis
- PDC 5 – Sharing of structures, systems, and components
- PDC 6 – Monitoring, inspection, testing, and surveillance

System specific PDCs, as applicable, will be listed and assessed in the subsections containing the system specific evaluations. Based on the information provided in the system specific subsections of PSAR section 6.4 and the NRC staff evaluations provided in each of the subsections below, the NRC staff finds that the applicant's identification of applicable PDCs is acceptable for the preliminary design.

As discussed in section 5.3 of this SE with respect to PDCs and section 5.2 of this SE with respect to SRDCs, LME has limited the scope of PDCs and SRDCs to DBEs and DBAs in a way that is inconsistent with the LBEs relying on the SR SSCs. The safety impacts of this inconsistency are being addressed through the application of reliability and capability targets along with special treatments that consider the full scope of the LBEs in which the SSC performs a PRA safety function (PSF), as discussed in PSAR section 6.3. The system specific

reliability targets and SSC specific capability targets were not included in the CP application, but preliminary special treatments for the SSCs support ensuring the safety functions relied on can be performed adequately for the full scope of LBEs. The final PDCs applied to each safety-significant SSC will be evaluated at the OL application stage and if the inconsistency remains, NRC staff will confirm the safety aspects are addressed appropriately through the final reliability and capability targets and special treatments applied.

6.4.1 Reactor Building

PSAR section 6.4.1 describes the Reactor Building (RB). The RB is a group of independent above-grade structures comprising four SR steel-frame reactor support structures (RXSTs) that each support a reactor and associated components; four NSRST steel-frame steam generator support structures (SGSTs), each supporting a steam generator; and an NSRST shield structure (SST) surrounding, and structurally separate from, the steel frames.

System specific regulatory requirements are as follows:

- 10 CFR Part 50, Appendix S

System specific guidance for the evaluation of the RB is as follows:

- RG 1.28, Revision 5, "Quality Assurance Program Criteria (Design and Construction)." (ML17207A293)
- RG 1.29, Revision 6, "Seismic Design Classification for Nuclear Power Plants." (ML21155A003)
- RG 1.76, Revision 1, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants." (ML070360253)
- RG 1.142, Revision 3, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)." (ML20141L613)
- RG 1.160, Revision 4, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." (ML18220B281)
- RG 1.221, Revision 0, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants." (ML110940300)
- RG 1.243, Revision 0, "Safety-Related Steel Structures and Steel-Plate Composite (SC) Walls for other than Reactor Vessels and Containments." (ML21089A032)
- NUREG-0800, Standard Review Plan (SRP) Chapter 3, "Design of Structures, Components, Equipment, and Systems."

In addition to PDCs 1 through 6, the following PDC is applicable to the RB:

- PDC-RFDC 71 - Reactor Support Structure Design Basis

6.4.1.1 Technical Evaluation

The NRC staff reviewed PSAR Section 6.4.1, which provides the specific design features of the RB SSCs that are responsible for meeting the SRDCs, the RFDCs, fulfilling RSFs for the RB SSCs, and fulfilling PSFs that prevent challenges to the RB features that fulfill RSFs. The SR and NSRST structures are evaluated below in the following three groups:

- RXSTs and common SR foundation
- SST and shared NSRST foundation
- SGSTs

For each group of structures, NRC staff evaluate their safety significance, compliance with applicable PDCs, and safety functions.

6.4.1.1.1 Reactor Support Structures (RXSTs) and Common Safety-Related Foundation

The NRC staff reviewed PSAR section 6.4.1.1, which describes the RXSTs and common safety-related foundation. The RXSTs are composed of structural steel braced frames, designed as ordinary concentrically braced frames. The common foundation for the RXSTs is a cast-in-place reinforced concrete mat supported on drilled shaft deep foundation elements. PSAR figures 6.4.1-1 and 6.4.1-2 show, in part, the plan geometry and elevation, respectively, and the related dimensions for an RXST. PSAR figure 6.4.1-3 shows, in part, a plan view of the common SR foundation and related dimensions. PSAR section 3.7 addresses, in part, the design and analysis methods and the construction materials used for all major LMGS structures, including the RXSTs and common SR foundation. The staff's acceptability of the design and analyses methods and construction materials for the LMGS structures is addressed in section 3.7 of this SE.

Safety Significance

LME classified the RXSTs and the common foundation that supports them as SR based on the functions that the RXSTs and common foundation perform, including:

- Supporting the reactor pressure vessel (RPV) and the reactor cavity cooling system (RCCS), and the corresponding loads from their operation
- Resisting and absorbing postulated loads from internal and external hazards
- Supporting SSCs other than the RPV/RCSS that are required to be in close physical proximity to the RPV/RCSS

The RXSTs support performance of RSF 1.4.2, "Maintain reactor support geometry," as stated in PSAR section 6.4.1.1.1. By performing this function, the RXSTs and common SR foundation prevent breach of the RPV and loss of core geometry during events that challenge the structural support of the reactor modules (e.g., design basis seismic event). The function performed by the RXSTs and common foundation is both preventive and mitigative in that the function prevents a larger release from occurring by reducing the frequency and consequence during events that challenge the structural support of the reactor modules.

Compliance with Applicable PDCs

The NRC staff reviewed PSAR section 6.4.1, which describes how the design bases for the RB comply with PDCs 1 through 6, including the RXSTs and common SR foundation. Additionally, the staff reviewed PSAR section 6.4.1.1.1, which addresses compliance with PDC-RFDC 71.

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR table 6.4.1-1, which provides the list of codes and standards applicable to the preliminary structural design of the RXSTs and common SR foundation to ensure that the LMGS design meets PDC 1. The principal codes and standards include AISC N690, ACI 349, ASCE 4, and ASCE 43. Additional information on the use of these codes and standards is addressed in PSAR section 3.7 and evaluated in section 3.7 of this SE. The principal codes and standards mentioned above are used along with additional codes and standards listed in PSAR table 6.4.1-1. The staff notes that these codes and standards are nuclear codes and standards and are traditionally used for the analysis and design of SR structures. Further, LME uses these codes and standards consistent with the guidance in regulatory guidance documents, including RG 1.243, RG 1.142, and Sections 3.7.2 and 3.8.4 of NUREG-0800, and are therefore applicable and acceptable for use in the analysis and design of the RXSTs and common SR foundation.

The NRC staff also reviewed the PSAR for quality assurance treatment applicable to the RXSTs and common SR foundation. PSAR section 6.4.1 states that the quality assurance program applicable to the RXST and common SR foundation is described in PSAR section 8.1. PSAR section 8.1 describes that SR SSCs, including the RXSTs and common SR foundation, comply with the requirements in 10 CFR 50, Appendix B, and ASME NQA-1. The full scope of the LMGS quality assurance program description (QAPD), described in PSAR section 8.1, is applied to the RXSTs and common SR foundation. Moreover, and consistent with PSAR section 8.1, PSAR table 6.4.1-1 lists ASME NQA-1 as one of the applicable standards for the RXSTs and common SR foundation. ASME NQA-1 addresses, in part, quality assurance requirements for installation, inspection, and testing of structural concrete, structural steel, and foundations. ASME NQA-1 is acceptable to the staff as endorsed by RG 1.28. PSAR table 1.4-2 states full conformance with RG 1.28. The NRC staff evaluation of the LMGS QAPD is provided in the QAPD TR SE and section 8.1 of this SE.

Based on the description of LME's use of nuclear codes and standards endorsed in NRC regulatory guidance and the quality assurance program requirements described above, the NRC staff finds that LME's preliminary design bases for the RXSTs and common SR foundation eet the requirements of PDC 1. The NRC staff will review the detailed implementation of these codes and standards in the final design at the OL stage.

Design Basis for Protection against Natural Phenomena (PDC 2)

The NRC staff reviewed the seismic categorization and associated criteria described in PSAR section 3.7. PSAR table 3.7-1 states that the RXSTs and common SR foundation are categorized as Seismic Category I (SC-I). Further, PSAR section 3.7.2.2 describes that SC-I SSCs are designed for the design response spectra (DRS) corresponding to the SSE (shown in PSAR figure 3.7-1). The use of SC-I and the SSE in the design of the RXSTs and common SR foundation conform to RG 1.29, as described in PSAR table 1.4-2. Per PSAR table 1.4-2, the SC-I seismic design provisions from RG 1.29 are applied to SR SSCs and these provisions require that LMGS SR SSCs be designed to withstand the effects of the SSE and remain

functional. Additionally, per PSAR table 1.4-2, the SC-I provisions from RG 1.29 related to seismic interaction, dynamic analysis, and quality assurance, are applied to SR SSCs. PSAR section 3.7 covers, in part, details related to the seismic analysis and design criteria for SR structures, including consideration of SSI effects. The NRC staff's evaluation of LMGS's seismic analysis and design criteria is addressed in section 3.7 of this SE. As described in PSAR sections 3.7.6, LME is committed to following ASCE 4-16 and ASCE 43-19 with **[[** **]]**. Based on the application of SC-I provisions, including the seismic design to the SSE, **[[** **]]**, the conformance to RG 1.29, and the seismic analysis and design criteria in PSAR section 3.7, the NRC staff finds that the applicant's approach for protecting the RXSTs and common SR foundation is acceptable and meets the requirements of PDC 2 and Appendix S to Part 50 with respect to consideration of the effects from earthquakes.

The NRC staff reviewed LME's consideration of tornado and hurricane effects on the RXST and common SR foundation. The NRC staff notes that these effects are addressed for the RXSTs and common SR foundation based on the protection provided by the SST. The consideration of these effects for the SST is discussed below in section 6.4.1.1.2, "Shield Structure (SST) and Shared NSRST Foundation," of this SE. Regarding the consideration of flooding effects, PSAR table 6.1-1 and section 2.4.5.6 establish the bounding flooding hazard for the LMGS site, based on consideration of hurricane, tsunami, and seiche hazards, among others. Figure 2.4.5-27 shows the pressures associated with the bounding flooding hazard that are applicable to safety-significant buildings. PSAR section 6.4.1.1.1 describes the strategies that LME implements for protecting SSCs against ingress of water, including locating SR SSCs supported by the RXSTs at an elevation above the still water flood level, and ensuring integrity of structures by considering the flood loads in their design. Additional flood protection strategies are addressed in PSAR section 6.1.1.4 and evaluated in SE section 6.1.1. Based on consideration of flood induced loads as part of the design loads for structures, the NRC staff finds that LME's approach for protecting the RXSTs and common SR foundation from the flooding hazard is acceptable and meets the requirements of PDC 2.

PDC 2 also requires that the design basis of safety-significant SSCs shall reflect appropriate combinations of the effects of normal, anticipated operational occurrence, design basis event, and design basis accident conditions with the effects of the natural phenomena. PSAR sections 6.4.1 and 6.4.1.1.1 point to PSAR section 3.7 for the loading conditions and combinations applicable to the RXSTs and common SR foundation. Additionally, PSAR section 6.4.1 indicates that the loads and their respective combinations take into consideration the structure's location, safety classification, and construction material. Per the design criteria in PSAR section 3.7, in addition to considering the effects of natural phenomena, the design of the RXST and common SR foundation considers other effects and loads as described in PSAR sections 3.7.4 and 3.7.5, which address applicable design loads and load combinations, respectively. The NRC staff finds that the load and load combinations described in PSAR section 3.7 meet the requirements in PDC 2 because the loads include normal, severe environmental, extreme environmental, and abnormal loads, and they are combined in accordance with the applicable design codes and standards. The NRC staff evaluation of the loads and load combinations for LMGS structures is presented in section 3.7 of this SE.

Based on LME's approach for considering the effects from natural phenomena along with the effects from other loads as described above, the NRC staff finds that LME's preliminary design bases for the RXSTs and common SR foundation to protect against natural phenomena meet

the requirements of PDC 2 and Appendix S to Part 50. The NRC staff will review the final design against the requirements of PDC 2 and Appendix S to Part 50 at the OL application stage.

Fire Protection (PDC 3)

The NRC staff reviewed PSAR section 6.4.1, which describes that the RXSTs and common SR foundation are designed and fabricated of materials that are high-temperature-resistant, inherently non-combustible, and non-fire sustaining. Moreover, this section states that fixed combustible materials at the location of the RXSTs are negligible and transient combustible materials are administratively controlled, resulting in low combustible loading and limited fire hazard potential. In addition, as indicated in PSAR section 6.4.1, LME's fire protection strategies include the implementation of fire protection systems that are designed, installed, and tested in accordance with NFPA standards. The NRC staff reviewed LME's approach for meeting PDC 3 and found it to be consistent with LME's fire protection program (FPP) described in PSAR section 8.6. Specifically, PSAR section 8.6 describes FPP elements that are implemented in an integrated manner to meet the requirements of PDC 3 in minimizing the probability and effect of fires and explosions. These elements include, in part, implementation of engineered fire detection and suppression systems to control fires and the building and system design and layout, including appropriate use of fire-resistant assemblies such as fire barriers, fire doors, fire dampers, and fire penetration seals. LME's FPP and its implementation are further evaluated in section 8.6 of this SE.

As described above, the NRC staff reviewed the information provided in PSAR sections 6.4.1 and 8.6 and determined that the preliminary design bases for the RXSTs and common SR foundation meet the requirements of PDC 3. This conclusion is based on LME's use of construction materials that are high-temperature-resistant, inherently non-combustible, and non-fire sustaining, and LME's commitment to implementing fire protection systems and features as necessary at the OL stage, consistent with its FPP. NRC staff will review fire zones and combustible loading along with fire protection systems and features related to the RXSTs and common SR foundation at the OL application stage. The staff evaluation of LME's approach for meeting the requirements of PDC 3, as described above, are also applicable to the SST, shared NSRST foundation, and SGSTs discussed below.

Environmental and dynamic effects design bases (PDC 4)

The NRC staff reviewed PSAR section 6.4.1, which describes how environmental conditions are considered in the design of the RXSTs and common SR foundation. The NRC staff also reviewed the supporting information addressed in other PSAR sections as described below. PSAR sections 3.7.5.2 and 3.7.5.3 summarize the load combinations for SR steel structures and foundations, respectively, which include consideration of high energy line break (HELB) loads. Per PSAR section 6.1.1.7, the HELB hazards are characterized as a break of a pipe or piping system component containing a fluid at maximum operating temperature exceeding 200°F or maximum operating pressure exceeding 275 psig. Further, per PSAR section 3.7.4.14, the primary HELBs postulated are main steamline breaks and breaks or leaks in the helium pressure boundary. These postulated breaks result in several loads, including pressure, thermal, pipe reactions, jet impingement, and missile impact (including pipe whipping), generated by or during the break, all of which are accounted for in the design load combinations for the RXSTs and common SR foundation.

PSAR section 6.4.1 indicates that the RB structures, including the RXSTs and common SR foundation, are designed to remain within acceptable material limit states and deflection values when subjected to the loads associated with the applicable environmental conditions. Additionally, PSAR section 6.1.1.7 indicates that SR SSCs will be designed to withstand or be protected from the impact of HELB missile without loss of capability to perform their RSFs by physically separating the SR SSCs from the potential HELB sources, providing missile barriers, or demonstrating that the HELB missile does not adversely impact the ability of SR SSCs to perform their RSF. The NRC staff finds that LME's approach to account for HELB loads in the design of structures, as applicable, is consistent with the provisions from AISC N690 and ACI 349, as endorsed by RGs 1.243 and 1.142, respectively, and is therefore acceptable. Further, the staff finds that the additional strategies described above provide a reasonable approach for protecting against HELB missile impacts because they address physical features and controls that would prevent HELB missiles from impacting the SR SSCs. Regarding external missile impacts, the RXSTs and common SR foundation are protected from such missiles by the SST.

The NRC staff also reviewed LME's consideration of the temperature of the operating environment, described in PSAR section 6.4.1.1.1. LME addressed temperature related considerations in accordance with AISC N690, "Specification for Safety-Related Steel Structures for Nuclear Facilities." LME's preliminary assessment indicates that the maximum operating temperature for the RXSTs remains below 250°F, which is the AISC N690-defined threshold above which certain material properties of structural steel (e.g., yield stress, modulus of elasticity) may require reduction due to elevated temperature conditions. LME will confirm that the temperature for the RXSTs, under normal operating and accident conditions, remains below the temperature threshold of NB3.3 of AISC N690 and will address exceedances, if any, in accordance with the provisions of AISC N690 as part of the final design in support of the OL application. Based on LME's conformance with AISC N690 as endorsed by RG 1.243, and the design margin confirmed by the staff via review of supporting information during the audit, the staff finds LME's approach for addressing temperature effects at the CP stage acceptable. The staff will review LME's adherence to the provisions in AISC N690 for the final design at the OL application stage.

Based on the NRC staff's evaluation described above, the NRC staff finds that LME's preliminary design bases for the RXSTs and common SR foundation meet the requirements of PDC 4. The NRC staff will review the final design against the requirements of PDC 4 at the OL application stage.

Sharing of structures, systems, and components (PDC 5)

The NRC staff reviewed the RXSTs and common SR foundation to ensure that the LMGS design meets PDC 5. The RXSTs are four independent structures, one for each reactor module, and therefore not shared between the reactor modules. The proposed design of the RXSTs meets PDC 5 because they will not be shared between reactor modules.

PSAR sections 1.1.3 and 6.4.1 describe information related to the sharing of the common SR foundation. All four RXSTs (and the reactor modules supported on the RXSTs) are supported by a common SR reinforced concrete mat (also referred to as a pile cap) that rests on top of concrete piers.

Per PSAR section 6.4.1.1.1, the common SR foundation (along with the RXSTs)

- supports the RPVs and RCCS and the corresponding loads from their operation;
- resists and absorbs postulated loads from internal and external hazards; and
- supports SSCs other than the RPVs/RCSS that are required to be in close physical proximity to the RPVs/RCSS.

The pile cap transfers the demands from the SSCs that it supports into individual piers. The common SR foundation system (along with the RXSTs) is SC-I, designed for the DRS corresponding to the SSE, and designed to withstand all the loading conditions and combinations as outlined in PSAR section 3.7.

As described above, the design criteria for the common SR foundation takes into consideration the postulated loads from internal and external hazards and associated load combinations, which include the loads resulting from the sharing of the common SR foundation across multiple reactor modules. On this basis, the staff finds that the sharing of the common SR foundation is adequately considered in the design criteria for the common SR foundation and will not impair the functions of the reactor modules, and therefore meets the requirements of PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR section 6.4.1, which addresses LME's consideration of monitoring, inspection, testing, and surveillance criteria as part of the design bases for the RXSTs and common SR foundation. The RXSTs and common SR foundation are designed to permit appropriate monitoring, periodic inspection, testing, and/or surveillance in accordance with the maintenance program described in PSAR section 8.3 and are informed by the guidance in NUMARC 93-01, Revision 4F, as endorsed by RG 1.160. Such activities include visual examinations and structural integrity/health monitoring through temperature, seismic, and radiation measurements, among others. The staff's evaluation of the maintenance program is in section 8.3 of this SE. The NRC staff review determined that LME's approach described above is informed by the guidance in NUMARC 93-01 as endorsed by RG 1.160. Based on the RXSTs and common SR foundation being designed to permit appropriate monitoring, periodic inspection, testing, and/or surveillance, and the conduct of such activities in accordance with LME's maintenance program, the staff concludes that LME's preliminary design of the RXSTs and common SR foundation meets the requirements of PDC 6. The review of detailed maintenance program requirements associated with the final design of structures will be addressed at the OL application stage. LME's approach for meeting the requirements of PDC 6 and the staff's evaluation described above are also applicable to the SST, shared NSRST foundation, and SGSTs.

Reactor Support Structure Design Basis (PDC-RFDC 71)

The NRC staff reviewed the design bases for the RXSTs and common SR foundation, which are described in PSAR section 3.7. The design bases criteria include consideration of normal, severe environmental, extreme environmental, and abnormal loads and respective combinations of those loads in accordance with ANSI/AISC N690 and ACI 349, which are endorsed in RG 1.243 and RG 1.142 for the design of SR steel and concrete structures, respectively. Further, per PSAR section 3.7.6, LME's seismic design bases assign the [[ ]] for SC-I structures, including the RXSTs and common SR foundation, which limits the response of these structures to [[ ]] for the combination of nonseismic demand and seismic demands. On this basis, the staff finds that the RXSTs and

common SR foundation will provide adequate structural support to protect the reactor geometry and therefore meets the requirements of PDC-RFDC71. The NRC staff will review the final design against the requirements of PDC-RFDC71 at the OL application stage.

Safety Functions

As described in PSAR section 6.4.1.1.1, the RXSTs support performance of RSF 1.4.2, "Maintain reactor support geometry," through application of PDC-RFDC 71. Additionally, PSAR section 6.4.1.1.1 describes a system-level SRDC derived from PDC-RFDC 71 that is applied to the RXSTs to ensure the performance of RSF 1.4.2. This SRDC is as follows:

> SRDC for RSF 1.4.2.1 (identified in PSAR Table 6.1-2 as "Maintain reactor stress levels"). —The reactor support system and structure are designed to assure that stresses developed during DBEs and DBAs are acceptable such that all required safety functions can be performed.

LME's design approach for the RXSTs and common SR foundation described in PSAR sections 6.4.1.1.1 and 3.7 takes into consideration the postulated loads from normal, severe environmental, extreme environmental, and abnormal conditions and associated load combinations. in accordance with the applicable codes and standards. Further, LME's design approach confirms that the stresses associated with such loads and load combinations do not exceed the design capacity of the RXST and common SR foundation. Moreover, in considering the combination of non-seismic demand and seismic demand, LME's proposed design approach limits the response of the RXSTs and common SR foundation to essentially elastic behavior. During the audit, to gain a better understanding of the information described in the PSAR, the NRC staff reviewed preliminary design reports that showed that the design demands are bounded by the design capacity of the RXSTs and common SR foundation. Based on the above, the staff finds that LME's preliminary design of the RXSTs and common SR foundation meets the requirements of SRDC for RSF 1.4.2.1 and therefore will adequately support performance of RSF 1.4.2.

6.4.1.1.2 Shield Structure (SST) and Shared NSRST Foundation

The NRC staff reviewed PSAR section 6.4.1.2, which describes the SST and shared NSRST foundation. The SST is a precast reinforced concrete structure, constructed with multi-wall precast concrete (i.e., individual solid precast concrete panels with full wall thickness inter-connected together). The SST foundation is a shared concrete mat between the SST, SGST and the FHAB. The shared section under the SST and SGST is supported by drilled shafts, and the section under the FHAB is supported by driven piles. PSAR figures 6.4.1-5 and 6.4.1-6 show the plan view and elevation, respectively, and the related dimensions for the SST. PSAR figure 6.4.1-3 shows, in part, a plan view of the shared NSRST foundation and related dimensions. PSAR section 3.7 describes, in part, the analysis and design criteria and materials applicable to the SST and shared NSRST foundation. The staff's evaluation of the design and analyses methods and construction materials for the LMGS structures is addressed in section 3.7 of this SE.

Safety Significance

As described in PSAR sections 6.4.1.2 and 6.4.1.2.1, the SST provides radiation shielding for personnel and equipment outside of the SST and within the various compartments of the SST. The SST houses and protects several systems from external DBHLs. A list of SR SSCs located within the SST is presented in PSAR section 6.1.1. Further, PSAR section 6.4.1.2.1 indicates that the SST is designed such that its failure in response to an external event cannot lead to a DBE or AOO that impacts performance of an RSF.

PSAR section 5.1.1 addresses safety classification criteria and the associated basis applicable to the SST, including the implementation of design and analysis special treatments that achieve performance levels consistent with the SSCs being supported/protected by the SST. PSAR section 5.5.2 and table 5.5-21, describe that, while not modeled in the PRA at the CP stage, the SST is one of the SSCs that was identified as NSRST via IDPP . PSAR table 5.5-21 indicates that the PSF for the SST is to prevent challenges to SR SSCs that perform RSF 1, RSF 1.1, RSF 1.2, and RSF 1.4.

Additionally, PSAR table 5.5-13, "Shield Structure for DID [Defense-in-Depth]," describes a special safety function identified as RBDO (event tree gate, reactor building pressure relief system open & integrity maintained), which aims to prevent adverse impacts on the ability of the SR SSCs to perform RSFs 1.4.1, and 1.4.2 for DID. Per PSAR section 6.4.1.2.1, the SST is equipped with a pressure relief system to prevent structural damage from over-pressurization caused by an HELB anywhere within the helium pressure boundary or secondary-side piping. Passive components, such as blowout panels and/or rupture discs designed to open at a specific setpoint relief pressure, prevent over-pressurization of the SST.

Compliance with Applicable PDCs

PSAR section 6.4.1.2.1 indicates that PDCs 1 through 6 are applied to the design of safety-significant SSCs, including the SST.

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR table 6.4.1-2, which provides the list of codes and standards applicable to the preliminary structural design of the SST and shared NSRST foundation. The principal design code used in the design of the SST and shared NSRST foundation is ACI 318, which is a well-established code for commercial construction and critical infrastructure. The analysis and design of the SST also consider provisions from ASCE 4, ASCE 43, ACI 349, and ACI 550.1R and other codes and standards listed in Table 6.4.1-2. The NRC staff notes that ASCE 4, 43, and ACI 349 are nuclear codes and standards traditionally used for the analysis and design of nuclear structures. The use of provisions from nuclear codes and standards is a special treatment for the design of the NSRST SST and shared NSRST foundation, and include, for example, analyzing the SST and shared NSRST foundation in accordance with the provisions of ASCE 43 and 4, and the use of provisions from ACI 349-13 for impulsive and impactive loads such as hurricane and tornado missiles and HELB loads. Also, the design criteria for the SST and shared NSRST foundation include conservative seismic demands [redacted]]] relative to the site-specific GMRS at the LMGS site. ACI 550.1R provides guidance, in part, on designing connection systems in a precast concrete structure so that its structural performance is equivalent to that of a conventionally designed, cast-in-place, monolithic concrete structure. During the audit, the NRC staff confirmed that LME's approach for the SST includes confirmation of equivalent performance to that of cast-in-

place reinforced concrete system, under the loads and associated load combinations described in PSAR section 3.7, by the implementation of validated analysis, design, and construction techniques.

The NRC staff also reviewed the PSAR for quality assurance treatment applicable to the SST and shared NSRST foundation. PSAR section 6.4.1 indicates the applicability of the LMGS QAPD, described in PSAR section 8.1, to the SST and shared NSRST foundation. The SST and shared NSRST foundation are designed, fabricated, constructed, and tested in accordance with the quality assurance program described in Part III of the LMGS QAPD and in PSAR section 8.1. Additional special treatments related to quality assurance will be specified at the OL application stage. Application of Part III of the LMGS QAPD at the CP stage is sufficient as a preliminary special treatment to meet the quality assurance requirements under PDC 1 for the SST and shared NSRST foundation. This finding is based on the NRC staff's evaluation of the LMGS QAPD in the QAPD TR SE and in section 8.1 of this SE.

The NRC staff finds that LME's preliminary design bases for the SST and NSRST shared foundation meet the requirements of PDC 1 because their design bases include the use of well-established standards, with conservative determination of demands (e.g., seismic demands), the commercial standards used are supplemented as necessary with provisions from nuclear codes and standards, and the SST and NSRST shared foundation are within the scope of LME's QA program. The NRC staff will review the detailed implementation of these codes and standards in the final design at the OL stage.

Design Basis for Protection against Natural Phenomena (PDC 2)

The NRC staff reviewed the seismic categorization and associated criteria described in PSAR section 3.7. Per PSAR table 3.7-1, the SST is categorized as SC-IIA and is designed to withstand an SC-IIA event, namely the SC-IIA DRS described in PSAR section 3.7.3.3. The SC-IIA DRS is **[**[redacted]**]**, as discussed above. Further, the SC-IIA DRS is conservative relative to the site-specific GMRS at the LMGS site. The adequacy of the SC-IIA DRS; the methodology for combining the effects of a seismic event with other effects/loads; and other seismic criteria, including consideration of SSI effects, are evaluated in section 3.7 of this SE. Further, per PSAR section 6.4.1.2.1, the SST is designed in accordance with the seismic design basis for NSRST structures detailed in PSAR section 3.7. As stated in PSAR section 3.7.6, **[**[redacted]**]]**, the staff finds that LME's approach for protecting the SST and shared NSRST foundation is acceptable and meets the requirements of PDC 2 and Appendix S to Part 50 with respect to the consideration of the effects from earthquakes.

The NRC staff reviewed LME's consideration of tornado and hurricane effects on the SST. Per PSAR sections 3.7 and 6.4.1.2.1, the SST is designed to withstand tornado and hurricane forces and missile impacts. Additional considerations of tornado/hurricane effects are addressed in PSAR sections 3.7.4.10 and 3.7.4.11, and table 6.4.1-2. Per these PSAR sections and table, the tornado and hurricane effects and missiles are considered in the design of the SST in accordance with RG 1.76, Revision 1, RG 1.221, and ACI 349-13, which is applicable to SR

structures. Based on LME's use of RGs 1.76, RG 1.221, and ACI 349-13 in its consideration of tornado and hurricane effects on the SST, the NRC staff finds that LME's approach for protecting the SST from tornado and hurricane hazards is acceptable and meets the requirements of PDC 2. Regarding consideration of effects of flooding, PSAR table 6.1-1 and section 2.4.5.6 establish the bounding flooding hazard for the LMGS site, based on consideration of hurricane, tsunami, and seiche hazards, among others. Figure 2.4.5-27 shows the pressures associated with the bounding flooding hazard that are applicable to safety-significant buildings. Further, per PSAR section 6.4.1.2.1, structural integrity of the overall building under DBHL flooding conditions is ensured by considering flood loads in the design of the structure as described in section 3.7. Based on LME's consideration of loads induced by the flooding DBHL in accordance with the referenced codes and standards and load combinations described in PSAR section 3.7 and table 6.4.1-2, the staff finds that LME's approach for protecting the SST from the flooding hazard is acceptable and meets the requirements of PDC 2.

The NRC staff reviewed PSAR section 3.7 which describes the loads and load combinations applicable to NSRST structures, including the SST and shared NSRST foundation. Per the design criteria in PSAR section 3.7, in addition to considering the effects of natural phenomena, the design of the SST and shared NSRST foundation considers effects and loads as described in PSAR sections 3.7.4 and 3.7.5, which address applicable design loads and load combinations, respectively. The loads considered include normal, severe environmental, extreme environmental, and abnormal loads. The NRC staff finds that the load and load combinations described in PSAR section 3.7 meet the requirements in PDC 2 because the loads considered include normal, severe environmental, extreme environmental, and abnormal loads and they are combined in accordance with the applicable design codes and standards. The NRC staff's evaluation of the loads and load combinations for LMGS structures is in section 3.7 of this SE.

Based on LME's approach for considering the effects from natural phenomena along with the effects from other loads as described above, the NRC staff finds that LME's preliminary design bases for the SST and shared NSRST foundation protect these structures against natural phenomena and meet the requirements of PDC 2. The NRC staff will review the final design against the requirements of PDC 2 and Appendix S to Part 50 at the OL application stage.

Fire Protection (PDC 3)

The NRC staff evaluation of LME's approach for meeting the requirements of PDC 3 for the RXST and common SR foundation, described above, are also applicable to the SST and shared NSRST foundation. Therefore, consistent with the staff's findings regarding LME's approach for meeting the requirements of PDC 3 for the RXST and common SR foundation, the staff finds that the design bases for the SST and shared NSRST foundation meet the requirements of PDC 3. The NRC staff will review fire zones and combustible loading along with fire protection systems and features related to the SST and shared NSRST foundation at the OL application stage.

Environmental and dynamic effects design bases (PDC 4)

The NRC staff reviewed PSAR section 6.4.1 and other PSAR sections mentioned below that address how environmental conditions are considered in the design of the SST and shared

NSRST foundation. PSAR section 3.7.5.1 summarizes the load combinations for concrete structures, including the SST. Per PSAR section 3.7.5.1 the design criteria for the SST includes loads associated with HELB. PSAR sections 6.1.1.7 and 3.7.4.14 describe characterization of HELB loads and postulated HELB sources. Further, PSAR section 6.4.1 indicates that RB structures, including the SST and shared NSRST foundation, are designed to remain within acceptable material limit states and deflection values when subject to these loads and other loads associated with the applicable environmental conditions. The NRC staff finds that LME's approach to account for HELB loads, as applicable, in the design of the concrete structures, including the SST and shared NSRST foundation, is consistent with the provisions of ACI 349-13, as endorsed by RG 1.142, and therefore is acceptable.

The NRC staff also reviewed LME's consideration of temperatures associated with the applicable environmental conditions for the SST and shared NSRST foundation. PSAR section 3.7.4.6 indicates that the SST and shared NSRST foundation are subjected to a normal operating temperature of 149 °F which is controlled by the design function of the reactor core cooling system (RCCS). This temperature is bounded by the normal operation temperature limits in ACI 349-13 and therefore is acceptable. Further, PSAR section 6.4.1.2.2 describes LME's approach for addressing temperatures within the SST and shared NSRST foundation under accident conditions, which relies on passive moderation by the RCSS to maintain temperatures under the ACI 349-13 accident temperature limits or alternative limits established via testing. The provisions in ACI 349-13 permit exposure to higher temperatures than those addressed in the code, provided that the effects of higher temperatures on concrete strength are established by tests and such effects are incorporated into the design. The NRC staff review finds LME's approach to maintain concrete temperatures under the applicable limits in ACI 349-13 or to justify exposure to higher temperatures by testing, if applicable, to be consistent with the provisions in ACI 349-13. Based on LME's conformance with ACI 349-13, as endorsed by RG 1.142, and the design margin confirmed by the staff via review of supporting information during the audit (see discussion under Safety Functions below), the staff finds LME's approach for addressing temperature effects at the CP stage acceptable. The staff will review LME's adherence to the provisions in ACI 349-13 in the final design at the OL stage.

Based on the staff's evaluation described above, the staff finds that LME's preliminary design bases for the SST and shared NSRST foundation meet the requirements of PDC 4. The NRC staff will review the final design against the requirements of PDC 4 at the OL application stage.

Sharing of structures, systems, and components (PDC 5)

The NRC staff reviewed PSAR sections 1.1.3 and 6.4.1, which cover information related to the sharing of the SST and the shared NSRST foundation. Per PSAR section 6.4.1.2, the SST (including the shared NSRST foundation that supports it) houses and protects the four RPVs among other SSCs. Further, PSAR section 6.4.1.2.1 states that the SST is classified as NSRST for preventing challenges to SR SSCs that perform RSF 1.1, RSF 1.2, and RSF 1.4. To meet this function, the SST design criteria in PSAR section 3.7 takes into consideration the postulated loads from internal and external hazards and associated load combinations, including loading conditions imposed by the reactor modules. In addition, PSAR section 3.7.6.4.2 indicates that LME will use safety factors, quality control, and load tests of individual installed piers/piles to ensure that differential settlement is kept within safe tolerances as specified in the codes and standards referenced in PSAR section 3.7.6.4.2. Based on the above, the NRC staff finds that the sharing of the SST and the shared NSRST foundation is adequately considered in the

design criteria for these structures. Therefore, the NRC staff concludes that LME's design bases for the SST and shared NSRST foundation meet the requirements of PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR Section 6.4.1, which addresses LME's consideration of monitoring, inspection, testing, and surveillance criteria as part of the design bases for the SST and shared NSRST foundation. LME's approach for meeting the requirements of PDC 6 and the staff's evaluation described above for the RXST and common SR foundation are also applicable to the SST and shared NSRST foundation. Additionally for the SST, LME's approach for meeting the requirements of PDC 6 includes the performance of testing of the SST relief capability in accordance with its maintenance program described in PSAR section 8.3. The NRC staff's review of PSAR sections 6.4.1 and 8.3 determined that LME's approach is informed by the guidance in NUMARC 93-01, as endorsed by RG 1.160. Based on the above, the NRC staff concludes that LME's preliminary design of the SST and shared NSRST foundation meets the requirements of PDC 6. The review of detailed maintenance programs requirements associated with the final design of structures will be addressed at the OL application stage.

Safety Functions

PSAR section 6.4.1.2.1 indicates that the SST is classified as NSRST for preventing challenges to SR SSCs that perform RSF 1, RSF 1.1, RSF 1.2, and RSF 1.4. As discussed under PDCs 1, 2, and 4 above, LME's design approach for the SST and shared NSRST foundation takes into consideration the postulated loads from normal, severe environmental, extreme environmental, and abnormal conditions, and associated load combinations, in accordance with applicable codes and standards. Further, LME's approach implements enhancements over applicable commercial standards, applied as special treatments, that include the consideration of a design basis earthquake that is based on the design basis earthquake for SR structures, consideration of extreme hurricane and tornado loads and HELB loads consistent with the criteria for SR structures, and supplementation of the commercial codes and standards with provisions from nuclear codes and standards. Additionally, per PSAR section 3.7.6, SC-IIA structures, are designed to **[[** [redacted] **]]**. During the audit, to gain a better understanding of the information described in the PSAR, NRC staff reviewed preliminary design reports that showed that the design demands are bounded by the design capacity of the SST and shared NSRST foundation. Based on the above, the NRC staff finds that LME's design bases for the SST and shared NSRST foundation will achieve adequate structural integrity to support performance of the SST and shared NSRST foundation's PSF of preventing challenges to SR SSCs that perform RSF 1, RSF 1.1, RSF 1.2, and RSF 1.4.

Steam Generator Support Structures (SGSTs)

The NRC staff reviewed PSAR section 6.4.1.3, which describes the SGSTs. The SGSTs are composed of structural steel and are designed as ordinary concentrically braced frames. The SGSTs share the NSRST foundation of the SST and FHAB. PSAR figures 6.4.1-1 and 6.4.1-2 show, in part, the plan geometry and elevation, respectively, and the related dimensions for an SGST. PSAR figure 6.4.1-3 shows, in part, a plan view of the shared NSRST foundation and related dimensions. PSAR section 3.7 describes, in part, the analysis and design criteria and materials applicable to SGSTs.

Safety Significance

PSAR section 6.4.1.3.1 states that the SGSTs support the steam generators and helium circulators, and provide partial support to the feedwater and main steam piping at their respective nozzles. Similar to the SST, the SGSTs are among the SSCs identified as NSRST via the IDPP. As was the case for the SST, PSAR table 5.5-21 indicates that the PSF for the SGSTs is to prevent challenges to SR SSCs that perform RSF 1, RSF 1.1, RSF 1.2, and RSF 1.4.

Compliance with Applicable PDCs

PSAR Section 6.4.1.3.1 indicates that PDCs 1 through 6 are applied to the design of safety-significant SSCs, including the SGSTs.

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR table 6.4.1-3, which provides the list of codes and standards applicable to the preliminary structural design of the SGSTs. The design basis for the SGSTs, like the SST discussed above, includes the use of commercial standards (e.g., ANSI/AISC 360) with enhancements applied as special treatments, including a conservative seismic demand and use of provisions from nuclear standards such as ASCE 4, ASCE 43, and ANSI/AISC N690, among other codes and standards listed in PSAR table 6.4.1-3. Examples of supplemental provisions from nuclear codes and standards used in the design of the SGSTs include analyzing the SGSTs in accordance with the provisions of ASCE 43 and 4, the use of **[[** [redacted] **]],** and the use of provisions from ANSI/AISC N690 for impulsive and impactive loads such as HELB loads.

The NRC staff also reviewed the PSAR for quality assurance treatment applicable to the SGSTs. PSAR section 6.4.1 indicates the applicability of the LMGS QAPD, described in PSAR section 8.1 to the SGSTs. The SGSTs are designed, fabricated, constructed, and tested in accordance with the quality assurance program described in Part III of the LMGS QAPD and PSAR section 8.1. Additional special treatments related to quality assurance will be specified at the OL application stage. Application of Part III of the LMGS QAPD at the CP stage is sufficient as a preliminary special treatment to meet the quality assurance requirements under PDC 1 for the SGSTs. This finding is based on the NRC staff's evaluation of the LMGS QAPD in the QAPD TR SE and in section 8.1 of this SE.

The NRC staff finds that LME's preliminary design bases for the SGSTs meet the requirements of PDC 1 because their design bases includes the use of well-established standards, with conservative determination of demands (e.g., seismic demands), the commercial standards used are supplemented as necessary with provisions from nuclear codes and standards, and the SGSTs are within the scope of LME's QA program. The NRC staff will review the detailed implementation of these codes and standards in the final design at the OL stage.

Design Basis for Protection against Natural Phenomena (PDC 2)

The NRC staff reviewed the seismic categorization and associated criteria described in PSAR section 3.7. Per PSAR table 3.7-1, the SGSTs are categorized as SC-IIA structures and are

designed to withstand an SC-IIA event, namely the SC-IIA DRS described in PSAR Section 3.7.3.3, [ ]. The adequacy of the SC-IIA DRS; the methodology for combining the effects of a seismic event with other effects/loads; and other seismic criteria, including consideration of SSI effects are evaluated in section 3.7 of this SE. Further, as SC-IIA structures, per PSAR section 3.7.6, the SGSTs are designed to meet [ ], the NRC staff finds that LME's approach for protecting the SGSTs is acceptable and meets the requirements of PDC 2 and Appendix S to Part 50 with respect to the consideration of the effects from earthquakes.

Regarding consideration of the effects of tornadoes and hurricanes, the staff notes that these are addressed for the SGSTs based on the protection provided by the SST. The consideration of these effects for the SST is discussed above. Regarding the consideration of effects of flooding, PSAR table 6.1-1 and section 2.4.5.6 establish the bounding flooding hazard for the LMGS site, based on consideration of hurricane, tsunami, and seiche hazards, among others. Figure 2.4.5-27 shows the pressures associated with the bounding flooding hazard that are applicable to safety-significant buildings. PSAR section 6.4.1.3.1 describes the flood protection strategies applicable to the SGSTs. The NRC staff review finds these strategies to be consistent with those applicable to the SR RXSTs and on this basis the staff finds that the flood protection strategies for the SGSTs are acceptable and meet the requirements of PDC 2 with respect to considering the effects of the flooding hazard at the LMGS site.

The NRC staff reviewed PSAR section 3.7, which describes the loads and load combinations applicable to NSRST structures, including the SGSTs. Per the design criteria in PSAR section 3.7, in addition to considering the effects of natural phenomena, the design of the SGSTs considers other effects in combination with the effects from natural phenomena, including effects from normal, severe environmental, extreme environmental, and abnormal loads.

Based on LME's approach for considering the effects from natural phenomena along with the effects from other loads as described above, the NRC staff finds that LME's preliminary design bases for the SGSTs to protect against natural phenomena meet the requirements of PDC 2. The NRC staff will review the final design against the requirements of PDC 2 and Appendix S to Part 50 at the OL application stage.

Fire Protection (PDC 3)

The NRC staff evaluation of LME's approach for meeting the requirements of PDC 3 for the RXST and common SR foundation, described above, are also applicable to the SGSTs because LME's approach for the SGSTs is consistent with the approach for the RXSTs and common SR foundation including the use of high-temperature-resistant, inherently noncombustible, and non-fire sustaining; the SGSTs are located in areas where fixed combustible materials are negligible and transient combustibles are administratively controlled; and implementation of fire protection systems that are designed, installed, and tested in accordance with NFPA standards. Therefore, the staff finds that the SGSTs meet the requirements of PDC 3. The NRC staff will review fire zones and combustible loading along with fire protection systems and features related to the SGSTs at the OL application stage.

Environmental and dynamic effects design bases (PDC 4)

The NRC staff reviewed PSAR section 6.4.1 and other PSAR sections mentioned below that address how environmental conditions are considered in the design of the SGSTs. PSAR section 3.7.5.2 summarizes the load combinations for SR steel structures, including those applicable to the SGSTs that address loads associated with HELB. PSAR sections 6.1.1.7 and 3.7.4.14 describe characterization of HELB loads and postulated HELB sources. Further, PSAR section 6.4.1 indicates that RB structures, including the SGSTs, are designed to remain within acceptable material limit states and deflection values when subject to these loads and other loads associated with the applicable environmental conditions. The NRC staff finds that LME's approach to account for HELB loads, as applicable, in the design of the steel structures, including the SGSTs, is consistent with the provisions of AISC N690, as endorsed by RG 1.243, and therefore acceptable.

Additionally, PSAR section 6.4.1 indicates that RB structures, including the SGSTs, are designed to withstand additional environmental effects such as corrosion and fire damage through adhering to relevant practices contained within referenced design and construction standards. Per PSAR table 6.4.1-3, these include provisions for protecting steel structures against corrosion effects from AISC 360 and AISC N690. Similarly, these include relevant provisions from these standards for considering the effects of elevated temperatures associated with fire conditions or elevated temperatures associated with other environmental conditions, in the design of steel structures. Additional staff review of fire protection is addressed under the evaluation of conformance with PDC 3. Consistent with the staff's evaluation of how the SGSTs design bases conform to PDC 1, the NRC staff determined that the use of the corrosion protection and temperature related provisions from these standards (AISC 360 and AISC N690) are appropriate for the design of the SGSTs.

Based on the NRC staff's evaluation described above, the NRC staff finds that LME's design bases for the SGSTs meet the requirements of PDC 4. The NRC staff will review the final design against PDC 4 at the OL application stage.

Sharing of structures, systems, and components (PDC 5)

The NRC staff reviewed PSAR section 6.4.1.3 and related subsections that, in part, describe functions and design criteria for the SGSTs. The SGSTs are individual and separate steel structures that provide separate structural support for each steam generator and related components.

Based on the independence of each SGST (i.e., an individual and separate SGST for each steam generator and related components), the staff determined that the SGSTs are not shared between the reactor modules. On this basis the staff finds that the SGSTs are consistent with PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR section 6.4.1, which addresses LME's consideration of monitoring, inspection, testing, and surveillance criteria as part of the design bases for the SGSTs. LME's approach for meeting the requirements of PDC 6 and the NRC staff's evaluation described above for the RXST and common SR foundation, are also applicable to the SGSTs.

On this basis the staff finds that the SGSTs are consistent with PDC 6. The review of detailed maintenance program requirements associated with the final design of structures will be addressed at the OL application stage

Safety Functions

PSAR section 6.4.1.3.1 states that the SGSTs are classified as NSRST for preventing challenges to SR SSCs that perform RSF 1, RSF 1.1, RSF 1.2, and RSF 1.4. As discussed under PDCs 1, 2, and 4 above, LME's design approach for the SGSTs takes into consideration the postulated loads from normal, severe environmental, extreme environmental, and abnormal conditions, and associated load combinations in accordance with the applicable codes and standards. Further, LME's approach implements enhancements over applicable commercial standards with enhancements applied as special treatments. Examples include the consideration of a design basis earthquake that is based on the design basis earthquake for SR structures, HELB loads consistent with the criteria for SR structures, and supplementing the commercial codes and standards with provisions from nuclear codes and standards. Additionally, per PSAR section 3.7.6, SC-IIA structures, are designed to meet **[[** [redacted] **]]**. During the audit, to gain a better understanding of the information described in the PSAR, NRC staff reviewed preliminary design reports that showed that the design demands are bounded by the design capacity of the STGSTs. Based on the above, the NRC staff finds that LME's design bases for the SGSTs will achieve adequate structural integrity to support performance of the SGSTs PSF of preventing challenges to SR SSCs that perform RSF 1, RSF 1.1, RSF 1.2, and RSF 1.4.

Technical Specifications

The RB has been identified to have technical specifications based on the preliminary design and the associated preliminary safety analysis, with additional details provided in Part IV of the CP application, "Technical Specifications." The proposed list of preliminary systems subject to LCOs identified in Part IV of the CP application identifies the availability of nuclear island heating, ventilation, and air conditioning (NIHV) and reactor cavity cooling system (RCCS) to support RSF 1.4.2.

Based on the review of the information provided in the CP application, as discussed above, the NRC staff finds the information provided meets 10 CFR 50.34(a)(5). The staff's finding is based on the variables, conditions, or items that are likely to become subjects of technical specifications for the RB being identified. Consistent with 10 CFR 50.34(b)(6)(vi), the NRC staff will evaluate the proposed technical specifications when detailed design information is available at the OL application stage.

6.4.1.2 Conclusion

Based on the staff's review of PSAR sections 6.4.1 and 3.7, and other sections of the PSAR as described above, the NRC finds that the information provided in the PSAR provides reasonable assurance that LME's preliminary design of the RXSTs, SST, and SGSTs meet the requirements of 10 CFR 50.34(a)(3), 10 CFR 50.34(a)(4), 10 CFR 50.34(a)(5), and 10 CFR Part 50, Appendix S. This conclusion is based on:

- LME's identification of the PDCs applicable to the RXSTs, SST, SGSTs, common SR foundation, and shared NSRST foundation, including PDCs 1 through 6, and PDC-RFDC 71 applicable to the RXSTs and common SR foundation.
- LME's description of design features, design criteria, and special treatments in PSAR sections and 6.4.1 and 3.7 used in the design of these structures to meet the applicable PDCs and ensure performance of safety-significant functions.
- LME's descriptions of materials of construction, general arrangement, and approximate dimensions addressed in PSAR sections and 6.4.1 and 3.7.
- LME's design basis described in PSAR sections 3.7 and 6.4.1, including description of preliminary and panned analyses and associated safety margin for structures and foundations.
- The information provided in Part IV of the CP application, identifying preliminary RB conditions that are subjects of technical specifications.

6.4.2 Fuel

PSAR section 6.4.2 describes the fuel used for the LMGS design as TRi-structural ISOtropic (TRISO) coated uranium oxycarbide (UCO) particles encased in a graphite matrix and surrounded by a fuel-free graphite zone in pebble form and contains the following information:

- PSAR section 6.4.2.1, "Fuel Functions and Design Criteria," describes the RSFs and PDCs associated with the fuel.
- PSAR section 6.4.2.2, "Fuel Performance and Operation," describes the fuel loading during initial reactor start and operation.
- PSAR section 6.4.2.3, "Fuel Qualification, Testing, and Program," describes the Xe-100 UCO fuel qualification and references topical report, "Xe-100 Topical Report – TRISO-X Pebble Fuel Qualification Methodology," Rev. 3, June 2023 (ML23160A293) (hereafter referred to as FQ-TR) which was reviewed and approved in the NRC staff's safety evaluation, subject to three limitations (the NRC staff's safety evaluation is included in the FQ-TR at ML23160A293).

PSAR section 6.4.2 provides design information for the SR low-enriched uranium pebbles (LUPs) and high-assay low-enriched uranium pebbles (HUPs). The NST non-fuel start-up fuel free pebbles (SFFP) are described in PSAR section 1.1.4.

System specific regulatory requirements applicable to the fuel are as follows:

- 10 CFR 50.34(a)(8) requires, for those SSCs that require research and development to confirm the adequacy of their design, an identification and description of the research and development program which will be conducted to resolve any safety questions associated with such structures, systems or components; and a schedule of the research and development program showing that such safety questions will be resolved at or before the latest date stated in the application for completion of construction of the facility.

System specific guidance applicable to the fuel includes:

- NUREG-2246, "Fuel Qualification for Advanced Reactors" (ML22063A131)

In addition to PDCs 1 through 6, the following system specific PDCs are applicable to the fuel:

- PDC 10 – Reactor Design
- PDC-Required Functional Design Criterion (RFDC) 11 – Reactor inherent protection
- PDC-RFDC 16 – Functional containment design
- PDC 28 – Reactivity Limits
- PDC-RFDC 34 – Residual heat removal
- PDC 62 – Prevention of Criticality in Fuel Storage and Handling

6.4.2.1 Technical Evaluation

Safety Significance

The fuel design plays a significant role in safety for the Xe-100 reactor due to its importance as the primary functional containment barrier for radionuclide retention, its role in heat removal, and the Xe-100 reliance on inherent reactivity feedback as a primary shutdown mechanism.

The performance of the fuel as the radionuclide retention barrier reduces reliance on other functional containment mechanisms such as confinement or containment barriers around the reactor which traditionally require active detection and isolation or filtration, which has varying levels of effectiveness depending on the radionuclides present. Alternatively, the functional containment barrier provided by TRISO fuel provides robust passive layers of protection that have historically been shown to remain effective over a wide range of temperature. The primary challenge to this radionuclide retention barrier comes in the form of temperature, shifting safety review focus from traditional detection and isolation to control of temperature through heat generation and heat removal.

The reliance on inherent reactivity feedback as a primary shutdown mechanism was evaluated based on a variety of factors. The first factor was that inherent reactivity feedback is not a function of the fuel alone. The reactivity of a system and its response to changes (i.e., reactivity feedback) is a function of the type of fuel, the arrangement of fuel, the material and arrangement of reflectors and moderators, the temperatures within the system, structural materials that impact absorption, etc. Reactivity is a system-wide characteristic and understanding the roles the fuel plays in that system and the impact that changes in the fuel can have on that system are key to understanding the safety significance of the fuel design relative to the inherent reactivity feedback of the system.

A second factor evaluated with respect to the inherent reactivity feedback was that inherent reactivity feedback cannot shut down the system or maintain the system in a low power or shutdown state unless the temperature of the system remains high (i.e., heat removal is restricted). This puts the fundamental safety function of controlling heat generation (i.e., reactivity and power) at odds with the fundamental safety function of controlling heat removal, which is generally thought of as maximizing heat removal. These key factors were evaluated along with other safety considerations important to the design of the fuel as part of this review.

Compliance with Applicable PDCs

As described in PSAR section 6.4.2.1, LME identified PDCs 1 through 6, 10, 28, and 62 and PDC-RFDC 11, 16, and 34 as applicable to the fuel.

Quality standards and records (PDC 1)

The full scope of the LMGS quality assurance program description (QAPD) described in PSAR section 8.1 is applied to the fuel. This program is designed to meet 10 CFR Part 50, Appendix B, and is evaluated through the QAPD TR SE and section 8.1 of this SE. Quality requirements related to fuel testing and qualification are also described in the "TRISO-X Pebble Fuel Qualification Methodology" TR, Revision 3 (ML23160A293), which is incorporated by reference into the PSAR in its entirety.

With respect to codes and standards applied, the NRC staff reviewed the preliminary fuel pebble product specifications in the audit to confirm that the standards and requirements applied are consistent with the preliminary design and analysis provided in the PSAR. The product specifications include ASTM standards to be used in testing key nuclear, mechanical, and thermo-physical properties. Mean values along with 95/95 and 95/98 tolerance bands were also specified for key parameters. The NRC staff found the scope of the specification and the values to be consistent with the preliminary design and analysis in the PSAR.

Based on this information, NRC staff finds that the preliminary fuel design meets PDC 1.

Design bases for protection against natural phenomenon (PDC 2)

The fuel is primarily protected from natural phenomenon by the systems that contain it and the facility structures those systems are contained within. These include the reactor pressure vessel (RPV), fuel handling system (FHS), reactor building, and the fuel handling annex building (FHAB) and spent fuel intermediate storage facility (SFSIF), which are evaluated in sections 6.4.6, 6.4.1, 7.3.6, and 7.3.1 of this SE, respectively. Additional information on design approaches to address the design basis hazard levels (DBHLs), including flooding and seismic, is also provided in PSAR sections 3.7 and 6.1 and evaluated in sections 3.7 and 6.1 of this SE. Unique external hazard aspects related to the fuel include core compaction in a seismic event and the potential for flooding to impact criticality safety. As described in PSAR section 3.6.12, the impact of core compaction in a seismic event was accounted for with a reactivity insertion in the seismic DBA. Potential criticality safety impacts from flooding were also addressed in PSAR section 3.9 and evaluated in section 3.9 of this SE. This included evaluations for optimal moderation in both the fresh fuel and spent fuel storage areas.

Based on this information, NRC staff finds that PDC 2 has been adequately addressed for fuel in the preliminary design.

Fire protection (PDC 3)

PSAR section 6.4.2 describes the fire protection strategies used where fuel is used, handled, or stored, including use of low combustible materials, limiting of transient combustibles, and the use of fire protection systems. The fuel itself is designed to withstand high temperatures without significant radionuclide release and the criticality safety analysis in PSAR section 3.9, which is evaluated in section 3.9 of this SE, included considerations for optimum moderation which

prevents potential safety issues that sprinklers could cause through the introduction of low-density hydrogenous material. Based on this information, NRC staff finds that PDC 3 has been adequately addressed for fuel in the preliminary design.

Environmental and dynamic effects design basis (PDC 4)

PSAR section 6.4.2 and the fuel qualification TR describe the ways in which the fuel is designed to withstand environmental and dynamic effects. This includes consideration of degradation mechanisms such as irradiation effects, oxidation, and erosion, along with the potential for mechanical damage such as dropping or crushing in the FHS. PSAR section 6.4.2.3 describes the additional fuel qualification test plans, which include testing of mechanical material properties such as crush strength and impact resistance and wear rate between fuel and other materials within the reactor core and FHS. The NRC staff finds these aspects of the preliminary fuel design and plans for testing sufficient for addressing PDC 4 at the CP stage. Final analysis of dynamic effects and the probability of dynamic effects damaging fuel will be provided at the OL application stage, as stated in PSAR section 6.4.2.

Sharing of structures, systems, and components (PDC 5)

Fuel is not shared between reactor modules. The extent of sharing between reactor modules with regard to the fuel is in the sharing of the foundation system supporting the reactor modules and the fuel handling systems. The sharing of these systems is evaluated in section 6.4.1 and 7.3.1 of this SE. The SFISF is also shared between reactor modules and is designed and evaluated with consideration for this sharing. For example, the limiting criticality safety analysis for the SFISF considered the offload of up to two cores concurrently when considering enrichment of stored fuel. The SFISF is evaluated in section 7.3.1 of this SE. Additional analyses related to criticality safety and fuel handling supplemental evaluations for the SFISF are evaluated in sections 3.9 and 3.10 of this SE, respectively.

Based on this information, NRC staff finds that PDC 5 has been adequately addressed for fuel in the preliminary design.

Monitoring, inspection, testing, and surveillance (PDC 6)

Fuel parameters are monitored through various remote online monitoring systems including the burnup measurement system (BUMS) and helium chemistry is monitored through the chemistry program described in PSAR section 8.8. Additionally, there are design features in the sphere unloading machines (SUMS) to separate damaged pebbles and remove them from circulation. NRC staff audited design details for the BUMS to confirm that the design supports this monitoring capability and the design facilitates the necessary calibration and replacement of the measurement systems without significant dose to workers. Based on this information, the staff found the preliminary design includes monitoring, inspection, testing, and surveillance functions for fuel that is consistent with the requirements of PDC 6. The staff will review the acceptability of the final design at the OL application stage.

Reactor Design (PDC 10)

PSAR section 6.4.2.1 states that, “Consistent with PDC 10, the fuel is designed with appropriate margin to ensure that the specified acceptable system radionuclide release limits [SARRDLs]

are not exceeded during any conditions of normal operation, including the effects of anticipated operational occurrences…" SARRDLs values were not included in the CP application, but NRC staff reviewed the PSAR dose consequence results for AOOs for a preliminary assessment, considering margin to the F-C target. The NRC staff found that of the 38 AOOs, 34 AOOs have a resulting mean dose consequence below 0.1 mrem. The AOO with the highest dose is SD-068, however, the consequence results specific to the SD-068 event were not available in the PSAR. To support the preliminary analysis, the long-term consequence from the DLOFC event was added for SD-068. The NRC staff found that because forced cooling is still available in SD-068, applying the DLOFC source term to this event, which is calculated based on only passive RCCS being available, is expected to be conservative. Based on this assessment, the NRC staff determined that the definition of specific SARRDL values can reasonably be left for later consideration, in accordance with 50.35(a)(2), and the staff has reasonable assurance that the applicant will be able to demonstrate the requirements of PDC 10 met at the OL application stage.

Reactor Inherent Protection (PDC-RFDC 11)

As noted earlier, the use of inherent reactivity as a credited design shutdown mechanism results in the two fundamental safety functions of controlling heat generation (i.e., limiting reactivity and power) and controlling heat removal (i.e., maximizing cooling) being at odds with each other. The NRC staff determined the calculations supporting the safety basis during the event timeframe that inherent reactivity is relied on must be performed with parameters at both extremes of the uncertainty bands to evaluate the safety margins related to both fundamental safety functions. This will narrow the operating range and require balance to ensure that safety limits are not reached on either side.

LME provided preliminary evaluations to assess the inherent reactivity feedback versus heat removal and maximum fuel temperature reached. These preliminary calculations indicate that the events analyzed in the PSAR represent the bounding conditions with respect to fuel temperature for these competing effects. As stated in section 5.2 of this SE, the NRC staff determined that this preliminary transient analysis performed by LME supported LME's position that the bounding fuel temperatures will occur when heat removal is most limiting and is adequate for assessing this safety aspect at the CP stage. Moreover, the results of the full scope transient analysis submitted in support of a future OL application, as discussed in PSAR section 3.1.7, will be confirmed by the NRC staff as part of the staff's review of the OL application.

As both the fuel and the moderator, the pebbles play an important role in achieving the overall inherent negative reactivity feedback being relied on in the LMGS design. LME provided preliminary temperature coefficient of reactivity results for the different components of the core in section 4.1 of the core design methods TR, evaluated in section 3.1.6 of this SE. The staff found preliminary results show that both the fuel and the moderator temperature coefficients are strongly negative.

Based on this information, NRC staff finds that the preliminary design for the fuel is consistent with the requirements of PDC-RFDC 11.

Functional Containment (PDC-RFDC 16)

PSAR section 6.4.2.1 states that, "The fuel consists of high-assay low enriched uranium (HALEU) or low-enriched uranium (LEU) uranium oxycarbide (UCO) TRi-Structural ISOtropic (TRISO) coated particles and pebbles that establish the primary functional containment barrier as described by PDC-RFDC 16." Additional details on the functional containment, and role of fuel in providing the principal barriers as part of the functional containment, is evaluated by NRC staff in section 3.1.4 of this SE. Details relative to fuel qualification and associated impact on functional containment are evaluated below. Based on the information provided in PSAR section 3.1.4, PSAR section 6.4.2, and the information incorporated by reference, the NRC staff finds that the fuel design is adequate to support PDC-RFDC 16 because (1) the mechanistic source term evaluation methodology is acceptable (see section 3.1.4 of this SE), and (2) LME has presented an acceptable fuel qualification plan (see evaluation below).

- Reactivity Limits (PDC 28)

PSAR section 6.4.2.1 states that, "Consistent with PDC-RFDC 11 and PDC 28, the fuel has an overall negative temperature coefficient of reactivity. This is due to Doppler broadening pertinent to the uranium kernel content and selection of the fuel pebble graphite and reflector graphite, which ensures stability of operations and negative reactivity insertion during core heat-up events." In section 3.1 of this SE, the NRC staff found that the DBA analysis methodology, thermodynamic analysis methodology, and core design methodology are acceptable for the preliminary analysis of the LMGS design. As discussed under PDC-RFDC 11, preliminary results for the reactivity coefficients showed that they are strongly negative, supporting that the rate of negative reactivity insertion will be sufficient to meet PDC 28.

Based on the evaluations performed in PSAR chapter 3 and the commitments in PSAR section 3.1.7 to provide additional transient analyses at the OL application stage, NRC staff finds that PDC 28 has been adequately addressed for fuel at the CP stage.

Residual Heat Removal (PDC-RFDC 34)

The NRC staff found the fuel design supports passive heat removal from the TRISO kernels to the helium coolant. As stated in PSAR section 6.4.2, key fuel design characteristics such as low power density, high thermal conductivity, and pebble size support the ability of the fuel to perform this function. The thermodynamic properties of the fuel will be confirmed through the fuel qualification described in PSAR section 6.4.2.3, which is part of the LMGS R&D program, as described in PSAR section 13.3.3. Preliminary fuel specifications for both the ILUPs and HUPs include requirements for key aspects that impact these parameters and ASTM standards to be used in performing the testing.

Based on this information, NRC staff finds that PDC-RFDC 34 has been adequately addressed for fuel in the preliminary design.

Prevention of Criticality in Fuel Storage and Handling (PDC 62)

PSAR Section 6.4.2.1 states that, "Consistent with PDC 62, the fuel is designed to ensure that new, used, and spent fuel pebbles in any potential configuration or handling operation within the FHS remain subcritical under any credible operating modes and conditions. Criticality safety for areas outside the reactor is further described in Section 3.9." The NRC staff's review of the requirements under PDC 62 are provided in section 3.9 of this SE. Staff confirmed that the

pebble parameters used in the PSAR section 3.9 analysis are consistent with the design description provided in PSAR section 6.4.2.

Transition from Non-fueled to Steady-State Core

Three fuel pebble types are utilized in the initial operation of the Xe-100, prior to it reaching steady-state from an online refueling perspective. The initial loading involves non-fuel start-up fuel free pebbles (SFFPs) and low-enriched uranium pebbles (LUPs). SFFPs contain no uranium or other fissionable material, are NST, and are described in Section 1.1.4.1.2 of the PSAR. LUPs are identical to the HUPs used in steady-state refueling, but contain uranium enriched to **[[** [redacted] **]]** versus the HUP uranium, which is enriched to 15.5%. Initially the core is filled with SFFPs. Fuel loading then begins with LUPs in a **[[** [redacted] **]]** of SFFP to LUP. The SFFPs loaded during this fill step are those returned from the bottom of the core. Criticality is reached when approximately **[[** [redacted] **]]** LUPs have been loaded into the core or when the core is approximately **[[** [redacted] **]]**.

This initial loading is performed with the RCSS control rods fully withdrawn to the bottom of the top graphite reflector with active neutron monitoring. The initial core loading is performed using a 1/M approach to critical, with a conservative approach to critical ensured with a stepwise approach. As the core reaches critical, the rods are inserted at a rate that compensates for the excess reactivity in the core while also allowing the core to heat and slowly increase in power as the LUP-SFFP mixture continues to be loaded into the core. **[** [redacted] **]** Before the reactor reaches full power a one-time transition to HUPs is made. As stated in PSAR section 6.4.2.2, procedures for initial startup will include hold points for physics testing.

The NRC staff reviewed preliminary analyses in audit, showing the planned transition points in the fill process, the hold points where physics testing will be performed, and preliminary results for the physics parameters throughout this transition of the core. One aspect discussed during audit was considerations for detector and startup source placement to ensure that the 1/M approach is conservative. LME plans to use a fixed source located in the reflector region, but final source type and placement have not been determined at this preliminary design stage.

The NRC staff determined that LME's plan for transitioning from a non-fuel core to the steady-state core, as described in PSAR section 6.4.2.2, is reasonable. Key aspects that help ensure safety during this process are the use of a conservative 1/M approach, the use of multiple hold-points during the transition to measure key physics parameters, and the single transition between LUP and HUP pebbles to minimize the risk that the incorrect pebble type would be loaded into the reactor. The NRC staff will confirm that the 1/M approach was implemented in a conservative manner when startup source and detector placement information is available at the OL application stage.

Fuel Qualification and Testing

PSAR section 6.4.2.3, "Fuel Qualification, Testing, and Programs," states that, "The Xe-100 UCO fuel qualification builds upon United States and international operating experience with TRISO fuel particles. The equilibrium core performance envelope of the Xe-100 fuel is based on the Department of Energy's (DOE) Advanced Gas Reactor (AGR) program." Additionally, PSAR Section 6.4.2.3 references FQ-TR (ML23160A293) for details on fuel qualification and provides a high-level summary of the X-energy Pebble Reactor Test (XPeRT) fuel irradiation campaign and includes dates for completing the test campaign. NRC staff compared these dates to the completion of construction date provided in PSAR Part I, "General and Financial Information," Section 1.4, "Construction Schedule," and determined that the irradiation campaign is scheduled to be completed before completion of construction because the test campaign is scheduled to be completed in Fall 2030 and completion of construction is scheduled to be completed in 2033. Completion of fuel qualification has been identified as part of the R&D program in PSAR section 13.3.3, which is evaluated in chapter 13 of this SE. Limitations and conditions on the FQ-TR are addressed by LME in PSAR table 1.4-5 and evaluated below in table 6.4.2-1. Based on the scope of LME's fuel qualification testing described in PSAR section 6.4.2.3 and the scope of R&D activities described in PSAR section 13.3.3 and evaluated in section 13.1.2 of this SE, along with the evaluation in Table 6.4.2-1 below, the NRC staff finds that fuel qualification and testing which requires research and development have been described sufficiently by the applicant, in accordance with 10 CFR 50.35(a)(3).

Table 6.4.2-1. Evaluation of Limitations and Conditions for FQ-TR

Limitation or Condition	LME Methods of Addressing (from PSAR table 1.4-5)	NRC Staff Evaluation
Limitation 1: Applicants referencing the TR should provide a justification for releases from defect particles, contamination in the fuel matrix and diffusion of volatile radionuclide in addition to the intact particle failure fraction derived from the testing to be used in a future licensing application. In effect, this methodology forms a portion of the fuel qualification for the final reactor fuel form, but not the entirety of the fuel qualification process for the performance of the fuel (e.g., cover the aspects cited above or the more comprehensive list of fuel qualification areas in NUREG-2246).	Justification for releases from defect particles, contamination in the fuel matrix, and diffusion of volatile radionuclides is provided by analytical models validated with AGR-1 and AGR-2 data. The results from the X-energy irradiation and post-irradiation examination (PIE) campaign will provide additional validation data and confirm that the performance of particles in Xe-100 pebble fuel is consistent with particles in AGR compacts. Operational releases will be justified via leadout irradiation results while AOO and accident releases will be justified via post-irradiation safety testing, as discussed in PSAR section 6.4.2.3 Justification for releases from the diffusion of volatile radionuclides is based on historical and AGR data as discussed further in PSAR section 3.1.4 for release analyses. The diffusion coefficients will be refined based on planned heating tests, as discussed in PSAR section 6.4.2.3.	NRC staff evaluated source term constituents, including diffusion coefficients and releases from defect particles and diffusion in section 3.1.4 of this SE. Intermediate source term results from XSTERM reviewed in audit showed significant increases in radionuclide release due to diffusion at increased temperatures. Validation against XSTERM showed that releases for most isotopes were conservative, a limited number of isotopes releases were underpredicted even with a higher fuel failure fraction than was seen in the AGR results. This may indicate that the associated diffusion coefficients are non-conservative. As discussed in section 3.1.4, diffusion coefficients will be evaluated further at the OL stage when additional test data is available and the full scope of XSTERM V&V and uncertainty analysis is available for review. Additionally, the staff audited preliminary fuel specifications and identified that they provided requirements related to defect

		particles and contamination in the fuel matrix which could result in increased releases relative to intact particle failures, as discussed in Limitation 1. Based on the determinations made in section 3.1.4 of this SE that the preliminary source term analysis is acceptable for the CP, this limitation has been sufficiently addressed for the CP stage.
Limitation 2: This approval is limited to the plan to test TRISO fuel and the operational envelope (temperature, burnup, and environmental conditions) outlined in the TR. Applicants referencing the TR should provide a set of fuel specifications and demonstrate how fuel to be used in the reactor will meet these specifications. Future licensing submittals will be needed to qualify the fuel for the totality of the operational regime, including but not limited to transient accident conditions and mechanical effects to the extent that these are not incorporated within the scope of the TR.	Xe-100 reactor fuel specifications were developed for manufacturing the test fuel to have a geometry that satisfies geometric requirements for the irradiation test train and is representative of the Xe-100 fuel pebbles. The refined final specification based on the test result evaluations will be summarized in future revisions of the Xe-100 TRISO Pebble Fuel Qualification Methodology LTR, which will be submitted prior to the Operating License Application. Prior to any fabrication effort, a Manufacturing Quality Plan will be created to identify the methods to verify all specified properties of produced fuel product and the associated sampling statistics to apply quality controls to a produced fuel batch. The Quality Control/Quality Assurance procedures for manufacturing are discussed in PSAR Section 6.4.2.3.	With respect to the limitation's point on fuel specifications, NRC staff reviewed the preliminary fuel specifications for both the LUPs and HUPs in audit. As discussed above, the specifications included requirements for key parameters and ASTM standards for how the testing should be performed. Based on the preliminary information provided, this portion of the limitation has been sufficiently addressed at the CP stage. The NRC staff notes LME's commitment to provide the final specification in an update to the FQ-TR and a Manufacturing Quality Plan prior to the OL application. With respect to the limitation's point on future licensing submittals needed to qualify the fuel for the totality of the operational regime, as discussed above, future testing through X-energy's XPeRT and XPLoRE programs are intended to demonstrate the fuel is qualified to transient accident temperatures and under mechanical loading, respectively. This testing was identified as an R&D item which will be completed prior to the completion of construction, which is sufficient to address this portion of the limitation at the CP stage.
EPRI TRISO TR Limitation 1 (addressed under FQ-TR Lim 3): The scope of this TR applied only to the UCO TRISO particles themselves. How the final fuel form is qualified and any impacts of the fuel form or other influences of the specific reactor design beyond the fuel form on the holistic fuel performance (for instance, any uranium contamination in the compact material) is the responsibility of the vendor or designer referencing this TR.	Xe-100 reactor fuel particle analytical performance will be validated by the results of the XPeRT irradiation and post-irradiation campaign. Results from heating tests are intended to be used to confirm applicability of AGR coated particle performance to the Xe-100 specification (particles embedded in pebbles), as discussed in PSAR Section 6.4.2.3.	Through the source term review, NRC staff reviewed the model constituents in XSTERM and confirmed that uranium contamination is addressed in the model. Current analysis uses AGR data as the basis but as stated in PSAR table 1.4-5, these values will be confirmed through the planned fuel testing. NRC staff did identify during the review that a primary difference between the AGR compacts and the LMGS fuel is the fuel free outer zone on the Xe-100

		pebbles. This provides additional radionuclide retention due to the additional diffusion needed prior to release. The NRC staff's review confirmed that this was explicitly modeled in XSTERM and that preliminary results for the additional retention appear reasonable, but additional data from the XPeRT testing is needed to confirm appropriate treatment at the OL stage. Preliminary information provided and planned fuel testing are sufficient to address this limitation for the CP stage.
EPRI TRISO TR Limitation 2 (addressed under FQ-TR Lim 3): This TR applies only to UCO TRISO particles that fall within the ranges discussed in Section 5.3 of the TR. If an applicant chooses to use UO2/UC2 ratios or burnup values that differ meaningfully from those used in the AGR program, the applicant must provide a justification for how the burnup and carbon content ratios conform to the performance ranges discussed in Section 5.3 of the TR.	The AGR data in Table 5-2 from Section 5.3 of the EPRI TR did not have ranges. Nevertheless, Xe-100 UCO TRISO particles U:O and U:C ratios are defined by the same specification range used as technical bases for the AGR fuel specifications. Therefore, the Xe-100 fuel UO2/UC2 ratios do not differ meaningfully from those used in the AGR program. The Xe-100 fuel burnup values are outside the AGR range. Planned testing discussed in PSAR Section 6.4.2.3 will address the differences.	The NRC staff finds this limitation sufficiently addressed for the CP stage based on the consistent UO2/UC2 ratios between AGR and Xe-100 fuel and the fuel testing identified to address the differences in fuel burnup.
EPRI TRISO TR Condition 1 (addressed under FQ-TR Lim 3): An applicant or licensee referencing this TR must evaluate any discrepancies between their fuel particles and the TRISO particles used in the AGR program – specifically, reviewing the ranges specified in Table 5-6 for stress values to capture any effects from different kernel sizes to ensure the data in the TR remain applicable.	See PSAR table 1.4-5 for a detailed evaluation of the Xe-100 fuel particle parameters against the AGR TRISO particle parameters. A summary is provided here: Both AGR-1 and AGR-2 parameters are within Xe-100 specifications for buffer thickness, IPyC thickness, IPyC density, IPyC anisotropy (with conversion applied), SiC thickness, SiC density, and OPyC density. AGR-1 buffer density measurements are within Xe-100 buffer density specifications. SiC aspect ratios for AGR-2 fit within the Xe-100 specification, but AGR-1 did not report the relevant value. LME inferred based on reported particle aspect ratio that the AGR-1 values would also fall within the Xe-100 specification. The OPyC thickness for AGR-1 and -2 mean confidence extrema range is mostly within the Xe-100 design specification. The tensile stress metrics will be calculated for the Xe-100 fuel specifications and key performance parameters from Xe-100 fuel performance envelopes and	The NRC staff finds this condition sufficiently addressed for the CP stage based on the consistency in key parameters described in table 1.4-5 for this condition and the fuel testing identified to confirm the performance of Xe-100 fuel relative to the AGR fuel.

	compared to AGR-1 and AGR-2 test envelopes and tensile stress metrics.	
EPRI TRISO TR Condition 2 (addressed under FQ-TR Lim 3): The performance limits in Table 6-6 and Figure 6-30 of the TR are the results of different tests with distinct samples, not all of which had the maximum bounds occur during the same test. The test results include considerations for uncertainty discussed in Section 6.4 and 6.5 of the TR. Further, when failures may occur, the data supporting the TR provides empirical evidence of individual particle failure. Applicants referencing this TR must ensure that they either remain within the tested bounds or justify how their proposed operating conditions remain applicable, considering uncertainty in both the AGR test results as described in the TR and any analytical uncertainty resulting from the proposed analytical method.	Fuel performance in Xe-100 normal operating conditions and most AOOs, DBEs, and DBAs will be demonstrated using AGR data, however, those not bounded by AGR data will be addressed by planned in-pile testing. Fuel performance during LBE scenarios and some upper bounding AOOs will be demonstrated using post-irradiation examinations of the X-energy Pebble Reactor Test (XPeRT). Data uncertainties are being taken into account in the analyses testing results as part of the fuel qualification methodology. See PSAR Section 6.4.2.3 Fuel Qualification, Testing, and Programs for additional discussion on these planned tests. These test results will be summarized in a future revision of the TRISO-X Pebble Fuel Qualification Methodology LTR provided as part of the operating license application.	The NRC staff finds this condition sufficiently addressed for the CP stage based on the fuel testing identified to confirm the performance of Xe-100 fuel relative to the AGR fuel to be performed as part of the R&D program. The NRC staff notes LME's commitments to address data uncertainties as part of the fuel qualification methodology and the commitment to update the FQ-TR prior to the OL application.
EPRI TRISO TR Condition 3 (addressed under FQ-TR Lim 3): Data discussed in this TR does not consider the impacts of short-lived fission products beyond those captured in the gas phase during experiments. Any applicant or licensee referencing this TR must disposition the impacts, if any, of short-lived fission products on the safety analyses and operational dose considerations resulting from short-lived fission products, in addition to the data discussed in this TR.	All significant radionuclides identified as important to dose, both short-lived and long-lived, are considered in the events analyzed by safety analysis. The output from safety analysis is fed as input to the analyses which consider operational dose. Therefore, by extension, the operational dose analyses have also considered and dispositioned short-lived radionuclides.	As part of the source term review documented in section 3.1.4 of this SE, staff reviewed the list of radionuclides LME identified as important to dose. The staff found this list to be reasonable based on a comparison to historical HTGR and a confirmatory analysis performed using SCALE. Based on this confirmation, the staff determined this condition has been addressed.

6.4.2.2 Conclusion

Based on the NRC staff's evaluation of information provided in PSAR Section 6.4.2, and associated information in the PSAR as described above, the NRC staff finds that the LMGS fuel design meets the requirements under 10 CFR 50.34(a)(3), 10 CFR 50.34(a)(4), 10 CFR 50.34(a)(8), and 10 CFR 50.35(a)(3). This finding is based on the following:

- The applicant's identification of applicable PDCs is reasonable;
- The LMGS preliminary design provided is reasonable for meeting the PDCs, as assessed above;

- Fuel qualification and testing has been described and included as part of the R&D program; and
- The L&Cs related to the FQ-TR have been addressed for the CP stage.

6.4.3 Core Metallic Structures

PSAR section 6.4.3 describes the core metallic structures (CMS) inside the reactor pressure vessel (RPV). The CMS consists of three core barrel (CB) shells (i.e., upper, middle, and lower shells), top plate, base pedestal, and lateral supports. The support shells provide lateral support for graphite and fuel and maintain geometry of the graphite core assembly through **[[** **]]** The base pedestal provides vertical support to the CMS, graphite core assembly and fuel, permits primary coolant circulation in the RPV bottom head region, and provides a connection for fuel outlet piping. The top plate consists of a lid, access cover, a fuel inlet transition pipe, and a closure spring. The functions of the top plate are to allow access to the graphite core assembly and to provide connections for the fuel inlet pipe and control rods. **[[** **]]** transfer vertical loads on the CMS to the RPV. The CMS is laterally supported by the RPV through bumpers at the top of each CB shell section that makes even contact with the RPV inner wall surface.

As stated in PSAR Section 6.4.3.1, the CMS is a reactor core support component that supports the graphite core structure and fuel pebbles, maintains structural integrity and geometry of the fuel pebbles, maintains core graphite structure geometry for control rod insertion to maintain safe shutdown, and transfers sufficient heat via conduction, convection, and radiation from the fuel to the reactor vessel.

System specific regulatory requirements applicable to the CMS are as follows:

- 10 CFR 50.34(a)(5)

System specific regulatory guidance applicable to the CMS is as follows:

- Regulatory Guide 1.87, Revision 3, “Acceptability of ASME Code Section III, Division 5, 'High Temperature Reactors'” (ML25176A084)

- RG 1.26, Rev. 5, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants Staff Regulatory Guidance 1.” (ML16082A501)

- Regulatory Guide 1.29, Revision 6, “Seismic Design Classification for Nuclear Power Plants” (ML21155A003)

In addition to PDCs 1 through 6, the following system specific PDCs are applicable to CMS:

- PDC 10 – Reactor design
- PDC-RFDC and PDC-CDC 26 – Reactivity control system
- PDC-RFDC and PDC-CDC 34 – Residual heat removal
- PDC-RFDC 70 – Reactor structural design basis

6.4.3.1 Technical Evaluation

Safety Significance

Except for the top plate lid access cover, which is classified as NST, all CMS components are classified as SR based on the following safety functions that the CMS system performs:

- Support and maintain structural integrity of the graphite core and fuel pebbles
- Withstand stresses generated during DBEs and DBAs to maintain core geometry such that control rods can be inserted to maintain safe shutdown
- Transfers necessary amount of heat via conduction, convection, or radiation from the fuel to the reactor vessel to ensure that radionuclide release limits are not exceeded during DBEs and DBAs

PSAR Section 6.4.3.1 states the CMS supports performance of RSFs 1.1.2.1 (provide means to insert negative reactivity to ensure safe shutdown to maintain long-term subcriticality), 1.2.1.1 (transfer heat from fuel to pressure vessel to maintain passive core heat removal) and 1.4.1.2 (limit stress in CMS to acceptable levels to maintain the structural integrity of the CMS and thus maintain core geometry). The RSFs 1.1.2.1 and 1.2.1.1 that the CMS performs to prevent or mitigate specific LBEs are modeled in the PRA, with the results summarized in Table 5.4-5. Per PSAR Section 5.4, RSF 1.4.1.2 is not explicitly included in the PRA logical model. However, RSF 1.4.1.2 is both preventive and mitigative in nature in that it prevents a larger release from occurring by reducing the frequency and mitigates consequence during events that challenge the structural integrity of the CMS. In addition, the CMS also supports NSRST PSFs 1.1.3.1 (control core reactivity to preclude reactor trips by controlling reactivity with control rods) and 1.2.2.2 (maintain helium flow path to maintain core heat removal through active means).

Compliance with Applicable PDCs

The NRC staff evaluated how the CMS design features satisfy the applicable PDCs below.

Quality Standards and Records (PDC 1)

Per PSAR Tables 6.4.3-1 and 6.4.3-2, all the CMS components are designed to the 2023 Edition of ASME BPVC Section III, Division 5, Subsection HG, which is endorsed by the NRC staff in RG 1.87, Rev. 3. The NRC staff found that Subsection HG is appropriate for design and construction of the CMS because it is intended for the design of Class SM metallic core support structures. The NRC staff finds the materials of construction listed in PSAR Table 6.4.3-2 meet the design and normal service level conditions listed in PSAR Table 6.4.3-1 in accordance with ASME BPVC III.5.HG. LME performed construction code assessment using the methodology as described in PSAR Section 3.8 and anticipated that once Level A, B, C, and D loading conditions are identified, the base pedestal will be classified as operating under elevated-temperature service conditions and **[** [redacted] **]]**. Per PSAR Section 3.8.2, test proposals will be developed to support strategies for addressing

areas requiring code supplementation as part of the material qualification plan. PSAR Section 6.4.3 also states that extending the qualified lifetime of 2.25Cr-1Mo is identified as part of the R&D program in Section 13.3. Based on the information above, the NRC staff finds that the preliminary design of the CMS meets the requirements of PDC 1 on using and supplementing appropriate codes and standards commensurate with the safety significance of the component safety functions. In addition, the full scope of the LMGS QAPD, described in PSAR section 8.1, is applied to the CMS. This program is designed to meet 10 CFR Part 50, Appendix B, and is appropriate for meeting the quality assurance requirements under PDC 1 for the SR aspects of the CMS. NRC staff evaluation of the LMGS QAPD is provided in the QAPD TR SE and section 8.1 of this SE.

Based on the above staff evaluation, the staff finds that the preliminary design of the CMS meets the requirements of PDC 1.

Design Basis for Protection against Natural Phenomena (PDC 2)

The NRC staff reviewed PSAR Section 6.4.3, which states that the CMS is designed to withstand seismic events and achieve performance levels corresponding to its designated seismic category as shown in Table 3.7-2 and as discussed in PSAR section 3.7. As stated in PSAR Table 3.7-2, the SR portions of the CMS are categorized as seismic Category I to ensure they can perform the associated RSFs during and following a seismic event. In addition, SSCs that could prevent the CMS SSCs from performing their RSF due to spatial interaction (i.e., seismic II/I) are categorized as seismic Category IIB to prevent this interaction. The seismic design requirements associated with each category and the codes and standards applied are described in PSAR section 3.7 and evaluated in section 3.7 of this SE. The NRC staff finds this seismic categorization and the associated design approach to be sufficient to address seismic aspects of PDC 2 for the preliminary design stage. Additional information on the final seismic design and the SSCs identified as needing seismic category IIB categorization to prevent interaction with this system will be reviewed at the OL application stage as part of the NRC staff's review of the final design. PSAR Section 6.4.3 also states that the CMS SR components are protected against natural phenomena, including tornadoes, external flooding, and extreme climate conditions, as described in Section 3.7, by their location within the RPV housed in the SST and RXST. The NRC staff agrees that the CMS SR components are protected against natural phenomena.

Based on the evaluation above, the NRC staff finds that the CMS design meets the requirements of PDC 2.

Fire Protection (PDC 3)

The NRC staff reviewed PSAR Section 6.4.3, which states that the probability that the CMS system is exposed to fires and explosions is minimized because it is made of high-temperature-resistant metallic materials, located within the inert HPB environment, and interfaces with the GCA that is fabricated of solid graphite, which is a stable material that can withstand extremely high temperature without igniting.

Based on the above information, the NRC staff finds that the CMS design meets the requirements of PDC 3.

Environmental and dynamic effects design bases (PDC 4)

The NRC staff reviewed PSAR Section 6.4.3, which states that PDC 4 requirements on the design to accommodate the effects of and to be compatible with the environmental conditions are satisfied by performing the environmental compatibility assessment and developing the material qualification plan to mitigate the applicable material degradation modes identified from the material compatibility assessment, as described in PSAR Section 3.8. An assessment of irradiation effects on mechanical properties is included in the R&D program as evaluated in Section 13.3 of this SE. The NRC staff evaluated the environmental compatibility assessment method in Section 3.8 of this SE and found that it meets the requirements of PDC 4.

The NRC staff also reviewed PSAR Section 6.4.3 to assess whether it meets PDC 4 requirements of protecting the CMS system against dynamic effects. PSAR Section 6.4.3 identified potential sources of dynamic hazards to the CMS and stated that as the design matures, the credibility of these dynamic hazards will be systematically assessed and addressed. The credibility assessment, final design approach, supporting analyses, and demonstration that dynamic effects have been appropriately addressed will be provided in the FSAR. The NRC staff finds that the PDC 4 requirements of protecting the CMS system against dynamic effects are met at the CP stage by identifying potential sources of dynamic hazards and having a plan to systematically assess and address the dynamic hazards identified. The NRC staff will review the FSAR to confirm compliance of the CMS design with PDC 4 during a future OL application review.

Based on the above discussion, the NRC staff finds that the preliminary design of the CMS system meets the requirements of PDC 4. Staff will evaluate the maturation of the R&D program through the required R&D updates as well as the final design at the OL application stage.

Sharing of structures, systems, and components (PDC 5)

The NRC staff reviewed PSAR Section 6.4.3, which states that the CMS components are not shared between reactor modules. It also states that CMS components share a foundation system with multiple safety-significant SSCs. The adherence to PDC 5 within the context of a shared foundation system is described in PSAR Section 3.7.6.3 and the NRC staff found it acceptable (see Section 3.7 of this SER). Based on the discussion above, the NRC finds that the preliminary design of the CMS system meets the requirements of PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR Section 6.4.3, which states that periodic inspections of the SR CMS components are performed in accordance with the RIM program, which is evaluated in SE Section 8.2. The CMS design provides access for implementing remote preservice and inservice requirements. The NRC staff reviewed the identified potential monitoring and nondestructive examination (MANDE) requirements, including visual examination, SSC health monitoring through temperature and radiation measurements, and helium-side chemistry monitoring and finds them appropriate on the basis that these methods are capable of identifying the principal degradation mechanisms expected for the CMS materials and operating environment, thereby providing adequate assurance to maintain the functional capability. In addition, LME states that the preoperational testing, as described in PSAR Section 12.1.1, will be performed to provide a baseline for the inservice inspections. Based on the discussion above, the NRC staff finds that the preliminary design of the CMS meets the requirements of

PDC 6, with review of the RIM program reasonably being left to the OL application review stage as discussed in section 8.2 of this SE.

Reactor Design (PDC 10)

The NRC staff reviewed PSAR Section 6.4.3, which states that PDC 10 is a higher-level requirement that is met in the CMS system design via application of PDC-CDC 26 and PDC-CDC 34. The NRC staff finds that this statement is appropriate because it is consistent with the discussion in the Xe-100 PDC Topical Report, previously approved by the NRC staff. As discussed in the basis for PDC 10 in the Xe-100 PDC Topical Report, PDC 10 is a design limit that supports acceptance criteria in PDC-CDCs 26 and 34. PDC-CDCs 26 and 34 describe the requirements for the function of reactivity control and active heat removal to meet the SARRDL, respectively. The NRC staff notes that SARRDLs were not included in the CP application, but the NRC staff reviewed the PSAR dose consequence results for AOOs for a preliminary assessment, considering margin to the F-C target and determined PDC 10 requirements are satisfied by meeting the requirements in PDC-CDCs 26 and 34. The evaluation for compliance of the CMS design to PDC-CDCs 26 and 34 is performed below.

Reactivity Control System (PDC-RFDC 26 and PDC-CDC 26)

The NRC staff reviewed PSAR Section 6.4.10, which states that LMGS reactivity control and shutdown system (RCSS) consists of two separate rod banks, the reactivity control system (RCS) control rods and the reserve shutdown system (RSS) shutdown rods, which performs the reactivity control functions required in PDC-CDC 26 and PDC-RFDC 26, respectively. As stated in PSAR Section 6.4.3, the CMS top plate lid design includes 18 bored holes that provide paths for RCSS rod movements. The holes align with the GCA sleeved channels that facilitate reliable movement of the nine RCS control rods and nine RSS shutdown rods used to control reactivity, as described in PSAR Section 6.4.10. In addition, as stated in PSAR Section 6.4.4, LME has developed a preliminary control rod insertion evaluation method to support the determination of the amount of rod insertion into the GCA considering the displacement of the GCC components. During the audit, the NRC staff confirmed that LME will validate the control rod insertion method by tests and analyses using commercial software such as LS-DYNA. The NRC staff also confirmed that LME's tests and analysis will account for the deformation of the channels and change in the alignment between the holes on the CMS top plate and GCA sleeved channels due to temperature change, irradiation, and other external loadings (e.g., transient and seismic loads) in the rod insertion evaluation.

Based on the above information, the NRC staff finds that the preliminary design of the CMS, combined with the control rod insertion evaluation method that will be validated by tests and analyses, meets the requirements of PDC-RFDC and PDC-CDC 26. The NRC staff will review the control rod insertion evaluation results to confirm that the requirements of PDC-RFDC and PDC-CDC 26 are met at the OL application stage.

Residual Heat Removal (PDC-RFDC 34 and PDC-CDC 34)

The NRC staff reviewed PSAR Section 6.4.3, which states that the passive heat removal function is achieved via the heat transfer path from the core, through the GCA and CMS, and on

to the RPV wall, the reactor cavity, the RCCS, and finally to the environment. The CMS design provides structural support for the core internal component geometries and tolerances needed to ensure adequate passive heat transfer. In addition, PSAR Section 6.4.3 states that the CMS maintains the core internal structure and features that direct helium flow through and around the core, thus supporting the active heat transfer.

Based on the above information, the NRC staff finds that the preliminary design of the CMS meets the requirements of PDC-RFDC and PDC-CDC 34.

Reactor Structural Design Basis (PDC-RFDC 70)

The NRC staff reviewed PSAR Section 6.4.3, which describes the design features of the CMS system: 1) it provides vertical support to the graphite core structure and fuel pebbles through the CB base pedestal connected to the RPV bottom and the closure spring at the top lid, and 2) it provides lateral support to the core and fuel pebbles through the graphite lateral supports (i.e., [[ ]]), and transfers the lateral load to the RPV through bumpers at the top of CB shell sections, which makes even contact to the RPV inner wall surface. LME states that [[ ]]

As discussed in the evaluation for PDC-CDC 26 and PDC-RFDC 26, the NRC staff finds that the preliminary design of the CMS, combined with the control rod insertion evaluation method to ensure sufficient rod insertion, meets the requirements of PDC 26 to ensure the insertion of the control rods and shutdown rods.

Based on the evaluation for PDC 1, 2, 4, and 6, the NRC staff finds that there is adequate assurance that the structural integrity of the CMS will be maintained for all the DBEs and DBAs.

The NRC staff finds that the preliminary design of the CMS, in addition to the ongoing design development in graphite lateral supports and the control rod insertion assessment method to ensure that the GCA maintains its geometry and safety functions, provides adequate assurance that the final design will meet the requirements of PDC-RFDC 70 to maintain the core geometry for sufficient heat removal and rod insertion. The staff will review the final design and analyses to confirm successful rod insertion and adequate lateral support to the GCAs from the CMS during all the LBEs at the OL application stage.

Safety Functions

PSAR Section 6.4.3 states the CMS supports performance of RSFs 1.1.2.1 (Provide means to insert negative reactivity to ensure safe shutdown to maintain long-term subcriticality), 1.2.1.1 (Transfer heat from fuel to pressure vessel to maintain passive core heat removal), and 1.4.1.2 (Limit stress in CMS to acceptable levels to maintain the structural integrity of the CMS and thus maintain core geometry). LME identified PDC-RFDC 26 (Reactivity control system), PDC-RFDC 34 (Residual heat removal), and PDC-RFDC 70 (reactor vessel and reactor system structural

design basis) as applicable design criteria for the CMS to support RSFs 1.1.2.1, 1.2.1.1, and 1.4.1.2. The CMS system also supports NSRST PSFs 1.1.3.1 (control core reactivity to preclude reactor trips by controlling reactivity with control rods) and 1.2.2.2 (maintain helium flow path to maintain core heat removal through active means). LME identified PDC-CDC 26 (Reactivity control system) and PDC-CDC 34 (Residual heat removal) to support these PSFs. In addition, LME identified PDCs 1 through 6 and PDC 10, with which the CMS design complies, as evaluated above. The NRC staff finds that these PDC selections align with the RSFs and NSRST PSFs identified for the preliminary design of CMS because they are consistent with PSAR tables 5.3-1 and 5.3-2, which map PDCs with RSFs and PSFs based on PRA results. The NRC staff finds that the CMS design features satisfy the required PDCs as evaluated in Section 6.4.4.3.2.

Based on the evaluation above, the NRC staff finds that the preliminary design of the CMS provides adequate support for RSFs 1.1.2.1, 1.2.1.1 and 1.4.1.2, as well as PSFs 1.1.3.1 and 1.2.2.2.

Technical Specifications

Part IV of the CP application identified CMS as being included in technical specifications based on the preliminary design and the associated preliminary safety analysis. The preliminary proposed design features include that the CMS is designed to support transferring heat from GCA to RPV (RSF 1.2.1.1), limit stress in CMS (RSF 1.4.1.2), and provide sufficient margin for safe shutdown.

The NRC staff finds the information provided in the PSAR for the CMS complies with 10 CFR 50.34(a)(5). The staff's finding is based on the variables, conditions, or items that are likely to become subjects of technical specifications for the CMS being identified in Part IV of the CP application. Consistent with 10 CFR 50.34(b)(6)(vi), the NRC staff will evaluate the proposed technical specifications when detailed design information is available at the OL application stage.

6.4.3.2 Conclusion

The NRC staff concludes that the preliminary CMS design is acceptable and meets the requirements of 10 CFR 50.34(a)(3)(i), 10 CFR 50.34(a)(3)(ii),10 CFR 50.34(a)(3)(iii), and 10 CFR 50.34(a)(5). This conclusion is based on the following:

- Based on the information provided in PSAR section 6.4.3.1, the applicant has met 10 CFR 50.34(a)(3)(i) for the CMS by identifying PDCs applicable to the CMS and describing the preliminary design of the CMS.
- Based on the information provided in PSAR section 6.4.3.1 through 6.4.3.3, the applicant has met 10 CFR 50.34(a)(3)(ii) by describing how the preliminary CMS design meets the requirements of the identified PDCs.
- Based on the information provided in PSAR section 6.4.3.1 through 6.4.3.3, the applicant has met 10 CFR 50.34(a)(3)(iii) by providing information relative to materials of construction, general arrangement, and approximate dimensions in the CMS, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.

- Based on the information provided in Part IV of the CP application, the applicant has met 10 CFR 50.34(a)(5) by identifying probable subjects of technical specifications for the CMS.

6.4.4 Graphite Core Assembly

PSAR Section 6.4.4 describes the Graphite Core Assembly (GCA), which is constructed of individual Graphite Core Components (GCCs). The GCA provides: 1) neutron reflection and mechanical support for the fuel pebbles as they move through the core, 2) channels for control rods and for the helium coolant, 3) heat transfer from the fuel to the CMS.

System specific guidance for the evaluation of the GCA is as follows:

- Regulatory Guide 1.87, Revision 3, "Acceptability of ASME Code Section III, Division 5, 'High Temperature Reactors'" (ML25176A084)

- Interim Staff Guidance (ISG) DANU-ISG-2023-01, "Material Compatibility for non-Light Water Reactors" (ML22203A175)

In addition to PDCs 1 through 6, the following system specific PDCs are applicable to the GCA:

- PDC 10 - Reactor Design
- PDC-RFDC 26 and PDC-CDC 26 - Reactivity Control Systems
- PDC-RFDC 34 and PDC-CDC 34 - Residual Heat Removal
- PDC-RFDC 70 - Reactor Structural Design Basis

6.4.4.1 Technical Evaluation

To support the qualification and design of the GCA, LME submitted Topical Report (TR) "TR009380, Revision A, Xe-100 Graphite Core Assembly Material Qualification and Design Methodologies," ML26114A162 (GQDM). The NRC approved this TR, with conditions, in "Safety Evaluation of TR009380, Revision 2, XE-100 Graphite Core Assembly Material Qualification and Design Methodologies," ML26069A593 (SE-GQDM). The limitations and conditions in the GQDM are addressed by LME in PSAR Table 1.4-14 and evaluated Section 1.3.4 of this SE.

Safety Significance

The GCA provides the structure of the core, forming the required geometry for pebble and coolant flow and the control rod channels. The GCA acts as a neutron reflector, mechanical support for the fuel to guide the pebbles as they move through the core, and maintains channels that provide paths for the RCSS rod movements. The GCA also plays a critical role in the control of heat removal by both creating the pathway for coolant flow and providing a path to conduct heat from the fuel to the CMS. The GCA supports safety functions listed in Section 6.4.4.1 of the PSAR.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

The GCA is classified as safety-related. The GCA partially conforms to the 2023 edition of ASME BPVC Section III, Division 5, as endorsed with conditions in RG 1.87, Rev. 3. The GQDM TR describes the areas of partial conformance, and the NRC staff's evaluation of the GQDM TR is found in SE-GQDM. The NRC staff reviewed PSAR Table 1.4-14, in which LME provides the method of addressing the conditions placed on the use of GQDM, and found the methods acceptable for all conditions in the staff evaluation, as discussed in Section 1.3.4 of this SE. PSAR Section 6.4.4 also states that the material qualification plan for the GCA is part of the R&D program, which is reviewed in Section 13.1.2 of this SE. The full scope of the LMGS QAPD, described in PSAR section 8.1, is applied to the GCA. This program is designed to meet 10 CFR Part 50, Appendix B, and is appropriate for meeting the quality assurance requirements under PDC 1 for the SR aspects of the GCA. The NRC staff's evaluation of the LMGS QAPD is provided in the QAPD TR SE and section 8.1 of this SE. Based on the discussion above, including the staff endorsement of ASME BPVC III.5 in RG 1.87, Rev. 3, the staff finds that the preliminary design of the GCA meets the requirements of PDC 1.

Design Basis for Protection against Natural Phenomena (PDC 2)

The GCA is not subject to the effects of tornadoes, hurricanes, floods, tsunami, and seiches because it is protected from these phenomena by the RPV, RXSTs, and SST within the RB. The GCA is designed to withstand seismic events and achieve performance levels corresponding to their designated seismic category as shown in PSAR Table 3.7-2 and discussed in PSAR section 3.7. The acceptability of seismic classification is documented in Section 3.7.3 of the staff SE. The PSAR states that hazard protection strategies, processes, and methods which are part of design considerations will incorporate more detailed deterministic and probabilistic hazard assessments at the OL application stage. Condition 17 in the SE-GQDM addresses the material properties of the irradiated graphite necessary for determining the seismic response of graphite to meet the requirements of PDC 2. Based on the discussion above, the NRC staff finds the information provided in the PSAR meets the requirements for PDC 2 at the CP stage.

Fire Protection (PDC 3)

Section 6.4.1.1 of the PSAR references the GQDM, which states the graphite used to fabricate the individual graphite core components (GCCs) to construct the GCA meet the requirements of ASTM D7219-08. Fabricating the graphite to this standard ensures the graphite is resistant to high-temperatures and is inherently noncombustible. The GCA is located within an inert helium pressure boundary, which precludes the initiation or propagation of fire. Fire protection systems deemed necessary by the fire hazards analysis and Post-Fire Safe-Shutdown Analysis (FSSA) are installed to protect the GCA. Therefore, the staff finds the GCA meets the requirements of PDC 3.

Environmental and dynamic effects design bases (PDC 4)

The GQDM describes the physical configuration, general design considerations, and environmental conditions for the GCA. Staff reviewed PSAR Table 1.4-14, in which LME

provides the method of addressing the conditions placed on the use of GQDM, and found the methods acceptable for all conditions in the staff evaluation in Section 1.3.4 of the SE. PSAR Section 6.4.4 also states that the material qualification plan for the GCA is part of the R&D program, which is reviewed in section 13.1.2 of the SE.

The radiation effects on graphite material properties were accounted for in the Flownex and GOTHIC models, as described in section 3.1.3 of the PSAR. As stated in section 3.1.3 of the PSAR, reflector graphite (i.e., GCA) thermal conductivity is modeled as a function of historical irradiation temperature, fast fluence, and transient temperature. As described in Sections 3.1.3.1 and 3.6.1.1 of this SE, the NRC staff reviewed this modeling in audit and performed confirmatory calculations to confirm, in part, that the thermal parameter adjustments for irradiation were reasonable. See Section 3.1.3.1 of this SE for the technical evaluation of the thermodynamic analysis and Section 3.6.1.1 for the MELCOR confirmatory calculations. Therefore, the staff finds the GCA meets the requirements of PDC 4.

Sharing of structures, systems, and components (PDC 5)

GCA SSCs are not shared between reactor modules. Therefore, the staff finds the GCA meets the requirements of PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

PSAR section 6.4.4 states that the GCA will meet the applicable portions of PDC 6 through a combination of preservice and inservice requirements. The NRC staff noted in PSAR Section 8.9 that the Comprehensive Vibration Assessment Program (CVAP), evaluated in SE Section 8.9, is applied to the GCA. The NRC staff determined that the application of CVAP to the GCA ensures identification, assessment, and mitigation of the potential adverse effects of flow, mechanically, and acoustically induced vibration on the GCA.

The GCA components are included in the RIM program, including ASME BPVC Code Case N-954-1, “Requirements for Monitoring and Nondestructive Examination (MANDE) of Graphite Core Components and a High-Temperature Gas Reactor,” which is evaluated in SE Section 8.2. The NRC staff review found that the GCA design will provide access for implementing remote preservice and inservice monitoring and examination. The NRC staff reviewed the preliminary MANDE requirements, including visual examination, SSC health monitoring through temperature and radiation measurements, and helium-side chemistry monitoring and finds them appropriate on the basis that these methods are capable of identifying the principal degradation mechanisms expected for the materials and operating environment, thereby providing adequate assurance to maintain functional capability of the GCA. In addition, LME states that the preoperational testing, described in PSAR Section 12.1.1.2, and startup testing, described in PSAR Section 12.1.1.3, will be performed to provide a baseline for the inservice inspections. Based on the discussion above, the NRC finds that the preliminary design of the GCA meets the requirements of PDC 6.

Reactor Design (PDC 10)

PSAR section 6.4.4 describes how the design of the GCA supports the requirements of PDC 10, including use of ASME 2023.III.5.HAB and ASME 2023.III.5.HHA, appropriate seismic and QA classifications, and graphite grades appropriate for the reactor environment. The NRC staff-approved GQDM TR describes these elements.

The PSAR states that PDC 10 is a higher-level requirement for reactivity control that is met via application of PDC-CDC 26 and, for heat removal, that is met via application of PDC-CDC 34. Based on the design of the GCA meeting the requirements of PDC-CDC 26 and 34, along with the staff evaluation in SE-GQDM, as conditioned, the NRC staff finds the GCA design meets the requirements of PDC 10 for the preliminary design of the GCA. Evaluations of PDC-CDC 26 and 34 are below.

Reactivity Control Systems (PDC-CDC 26 and PDC-RFDC 26)

PSAR section 6.4.4 states the GCA is designed to provide channels for the insertion path of the reactivity control and shutdown system rods. Acceptability of reactivity control is discussed in SE Section 6.4.10. The ability of the GCA to maintain geometry is found in the NRC staff-approved GQDM TR. In addition, PSAR section 6.4.4 discussed a preliminary control rod insertion evaluation method for the performance of control rod insertion into graphite reflector channels considering the vertical and lateral displacements of individual GCCs for different load conditions. LME states that this method will be used to adequately size GCCs and apply deformation limits to ensure sufficient margin to failure is maintained throughout the GCA operational life. During the audit, the NRC staff confirmed that LME will validate the control rod insertion method by tests and analyses using commercial software such as LS-DYNA. The NRC staff also confirmed that LME's tests and analyses will account for the deformation of the channels and change in the alignment between the holes on the CMS top plate and GCA sleeved channels due to temperature change, irradiation, and other external loadings (e.g., transient and seismic loads) in the rod insertion evaluation. Based on this information, the NRC staff found the methodology an acceptable approach to ensure margin to failure is maintained through design life.

The NRC staff determined the preliminary GCA design meets the requirements of PDC-CDC 26 and PDC-RFDC 26 because the insertion path for the reactivity control and shutdown system rods will be maintained to support reactivity control. The NRC staff will review the control rod insertion evaluation results to confirm that the requirements of PDC-CDC 26 and PDC-RFDC 26 are met at the OL application stage.

Residual Heat Removal (PDC-CDC 34)

PDC-CDC 34 requires an active means to transfer fission product decay heat and other residual heat from the reactor core to an ultimate heat sink at a rate such that specified acceptable system radionuclide release design limits are not exceeded during AOOs. The preliminary design supports active heat transfer through design features of the GCA including helium riser channels, core inlet and outlet plenums, and geometries which direct helium coolant through and around the core. Therefore, the staff finds the GCA meets the requirements of CDC-CDC 34.

Residual Heat Removal (PDC-RFDC 34)

PSAR section 6.4.4 states the GCA is designed to provide a heat transfer path from the core through the GCA by establishing specific geometries and tolerances to ensure adequate passive heat transfer during DBEs and DBAs. The ability to maintain geometry is supported by the NRC staff-approved GQDM TR, with conditions found in the SE for the TR. The NRC staff determined that the preliminary GCA design has a geometry and tolerances that will allow for residual heat removal from the core to the environment and meets the requirements of PDC-RFDC 34.

Reactor Structural Design Basis (PDC-RFDC 70)

PSAR section 6.4.4 states that the GCA is designed to maintain an acceptable core geometry. Key features of the geometry include single-column stacking to minimize tensile loading, connecting and shaping the GCCs to prevent excessive movement and stress concentrators, and utilizing two graphite grades to accommodate differences in temperature and fluence throughout the GCA. The area of the GCA with the highest neutron flux is designed to be replaceable to maintain integrity throughout the design life. Vertical channels for both control rods and helium risers are also design features to assist with GCC alignment. Additional information about material qualification and design is found in the NRC staff-approved GQDM TR. The NRC staff determined the preliminary GCA design meets the requirements of PDC-RFDC 70 as the geometry allows for passive removal of residual heat and permits insertion of neutron absorbers.

Safety Functions

PSAR Section 6.4.4 states that the GCA supports performance of RSFs 1.1.2.1 (Provide means to insert negative reactivity to ensure safe shutdown to maintain long-term subcriticality), 1.2.1.1 (Transfer heat from fuel to pressure vessel to maintain passive core heat removal), and 1.4.1.3 (Maintain acceptable GCA geometry). LME identified PDC-RFDC 26 (Reactivity control system), PDC-RFDC 34 (Residual heat removal) and PDC-RFDC 70 (reactor structural design basis) as applicable design criteria for the GCA to support RSFs 1, 1.1, 1.1.2, 1.2, 1.2.1, 1.2.1.1, 1.4, 1.4.1, and 1.4.1.3. The GCA also supports NSRST PSFs 1.1.3, 1.1.3.1, 1.2.2, and 1.2.2.2. LME identified PDC-CDC 26 (Reactivity control system) and PDC-CDC 34 (Residual heat removal) to support these PSFs. The NRC staff finds that these PDC selections are consistent with PSAR Section 5.3 and align with the RSFs and PSF identified for the GCA because they are consistent with PSAR tables 5.3-1 and 5.3-2, which map PDCs with RSFs and PSFs based on PRA results.

Based on the PDC and safety function evaluations described above, the NRC staff finds that the preliminary design of the GCA provides adequate support for RSFs 1, 1.1, 1.1.2, 1.2, 1.2.1, 1.2.1.1, 1.4, 1.4.1, and 1.4.1.3, and NSRST PSFs 1.1.3, 1.1.3.1, 1.2.2, and 1.2.2.2.

Technical Specifications

The GCA has been identified in technical specifications based on the preliminary design and the associated preliminary safety analysis, with additional details provided in Part IV of the CP application, "Technical Specifications." The preliminary proposed design features include that the GCA 1) controls reactivity with inherent reactivity feedback and provides sufficient shutdown margin for safe shutdown, 2) controls heat removal by passively transferring heat from the fuel to the ultimate heat sink, and 3) maintains core geometry by limiting core reflector graphite stress to acceptable levels to support RSFs 1.1.2.1, 1.2.1.1, and 1.4.1.3.

Based on the information provided in the PSAR, the staff finds the preliminary information regarding technical specifications for the GCA addresses the requirements of 10 CFR 50.36(c)(4), "Design Features."

In addition, the NRC staff finds the information provided meets the requirements of 10 CFR 50.34(a)(5). The staff's finding is based on the variables, conditions, or items that are likely to become subjects of technical specifications for the facility being identified in Part IV of the CP application. Consistent with 10 CFR 50.34(b)(6)(vi), the NRC staff will evaluate the proposed technical specifications when detailed design information is available at the OL application stage.

6.4.4.2 Conclusion

Based on the discussion above, the NRC staff concludes that the preliminary design for the GCA is acceptable and meets the requirements of 10 CFR 50.34(a)(3)(i), 10 CFR 50.34(a)(3)(ii), 10 CFR 50.34(a)(3)(iii), and 10 CFR 50.34(a)(5). This conclusion is based on the following:

- Based on the information provided in PSAR section 6.4.4.1, the applicant has met 10 CFR 50.34(a)(3)(i) for the GCA by identifying PDCs applicable to the GCA.
- Based on the information provided in PSAR section 6.4.4.1, the applicant has met 10 CFR 50.34(a)(3)(ii) by describing how the preliminary GCA design meets the PDCs identified for the GCA.
- Based on the information provided in PSAR section 6.4.4.1, the applicant has met 10 CFR 50.34(a)(3)(iii) by providing information relative to materials of construction, general arrangement, and approximate dimensions in the GCA, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.
- Based on the information provided in Part IV of the CP application, the applicant has met 10 CFR 50.34(a)(5) by identifying probable subjects of technical specifications for the CMS.

6.4.4.3 References

- Regulatory Guide 1.87, Revision 3, “Acceptability of ASME Code Section III, Division 5, 'High Temperature Reactors'” Accession Number: ML25176A084
- Interim Staff Guidance (ISG) DANU-ISG-2023-01, “Material Compatibility for non-Light Water Reactors”(ML22203A175).
- U.S. NRC Final Safety Evaluation for X-Energy Xe-100 Graphite Core Assembly Material Qualification and Design Methodologies TR, Revision 2, (ML26069A593).
- Submittal of X Energy, LLC (X-Energy) Xe-100 Licensing Topical Report Graphite Core Assembly Material Qualification and Design Methodologies, Revision 2 Accepted Version, (ML26114A162).
- ASME BPVC Code Case N-954-1, “Requirements for Monitoring and Nondestructive Examination (MANDE) of Graphite Core Components and a High-Temperature Gas Reactor.”

6.4.5 Defuel Chute

PSAR Section 6.4.5 describes the defuel chute (FUCH). The FUCH provides the path for fuel pebbles exiting the reactor core to enter the Fuel Handling System (FHS) via a portion of the defuel chute pressure vessel (DCPV). The DCPV surrounds the FUCH and provides the helium pressure boundary (HPB). The FUCH is assembled from two extruded metal cylinders connected by a sliding joint to allow for axial thermal expansion. The FUCH is sized with a diameter sufficient to prevent pebble bridging that obstructs pebble flow and with a length sufficient for decay of fuel pebbles such that fuel temperature at exit from the FUCH is compatible with the lower design temperature materials used for the DCPV and FHS. The FUCH design and service level conditions are listed in PSAR table 6.4.5.1.

System specific regulatory requirements applicable to the FUCH are as follows:

- 10 CFR 50.34(a)(5),
- 10 CFR 50.34(a)(8),
- 10 CFR 50.34(a)(12), and
- 10 CFR 50.35(a)

System specific guidance for the evaluation of the FUCH is as follows

- Interim Staff Guidance (ISG) DANU-ISG-2023-01 “Material Compatibility for non-Light Water Reactors,” (ML22203A175).
- RG 1.87, Rev. 3, “Acceptability of ASME Code Section III, Division 5, 'High Temperature Reactors'” (ML25176A084).

In addition to PDCs 1 through 6, the following PDCs are applicable to the FUCH:

- PDC 10 – Reactor Design
- PDC-RFDC34 and– Residual Heat Removal
- PDC 70 – Reactor Vessel and Reactor System Structural Design Basis

6.4.5.1 Technical Evaluation

Safety Significance

The primary function of the FUCH is to maintain the fuel geometry within the reactor core area such that reactivity can be controlled and heat removed (RSF 1.4.1.2 – withstand stresses developed during DBEs and DBAs and ensure acceptable reactor core geometry is maintained to control reactivity and core heat removal). It accomplishes this function by providing a channel for the movement of fuel pebbles from the reactor vessel fuel outlet funnel portion of the bottom reflector, which is a high-temperature graphite component, to the defuel chute pressure vessel (DCPV), which is a low-temperature metallic component. By performing this function, the FUCH supports the high-temperature fuel and prevents fuel contact with structures not designed for those temperatures (i.e., the reactor vessel lower head and DCPV). PSAR section 5.4 states that this function is not explicitly included in the PRA logical model. However, the support function is a passive structural function that will be subject to a comprehensive structural evaluation as outlined in PSAR section 3.8, research and development as outlined in PSAR section 13.3.2, and will be included in the PRA logic model at the OL application stage.

The temperature transition is managed by transferring heat from the fuel to a heat sink, controlling the heat generated from fuel decay within the FUCH, and preventing the addition of significant heat from fission. The heat transfer function includes both a passive heat transfer function (RSF 1.2.1.1 – transfer sufficient heat to the DCPV and RPV to assure that fuel and radionuclide release limits are not exceeded during DBEs and DBAs) and an active heat transfer function (PSF 1.2.2.2 – accommodate restoration of primary flow following various AOOs and ensure that SARRDLs are not exceeded). The heat transfer function also includes heat removal necessary to maintain the integrity of the FUCH (explained below under PDC 34), which is potentially susceptible to damage at temperatures lower than those that would challenge the fuel design (e.g., SARRDLs). The heat transfer function maintains the FUCH temperature within limits by removing the decay heat released by the fuel and the heat generated within the FUCH during reactor operation by direct gamma heating and fission. The geometry of the FUCH must ensure that the fuel within the FUCH is subcritical, which prevents sustained fission during transfer of fuel pebbles from the reactor core to the FHS.

Cooling of the FUCH without primary helium flow relies primarily on convective and radiative heat transfer from the FUCH to the surrounding vessels (i.e., the DCPV and the lower head of the RPV), which is supported by the indirect external cooling of these vessels by the reactor cavity cooling system (RCCS). The FUCH includes a design feature to support active cooling with cooled primary helium flow from the steam generators.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

For the FUCH, all components are classified as SR, constructed of Alloy 800H, and, consistent with the guidance in RG 1.87, Rev. 3, are designed to ASME BPVC Section III (2023), Division

5, Subsection HG, Subpart B. Subsection HG, Subpart B, is the appropriate subsection for construction of the FUCH because it is intended for Class A components performing a structural support function (i.e., non-pressure-retaining) in environments exceeding the temperature limits in Table HAA-1130-1. The FUCH is identified as part of the R&D program for SR metallic SSCs described in PSAR section 13.3.2 and table 13.3.2-1. The scope of the R&D program for addressing areas requiring Code supplementation for the FUCH includes extension of qualified life. The preliminary FUCH design temperature is within the allowed limits of ASME.III.5.HGB, but final thermal analysis has not been performed. If the design temperature established through incorporation of the FUCH required functions within the PRA logic model exceeds the allowable limits, the qualified temperature of Alloy 800H would need to be extended to support safety findings at the OL application stage.

The full scope of the LMGS QAPD, described in PSAR section 8.1, is applied to the FUCH. This program is designed to meet 10 CFR Part 50, Appendix B, and is appropriate for meeting the quality assurance requirements under PDC 1 for the SR aspects of the FUCH. The NRC staff's evaluation of the LMGS QAPD is provided in the QAPD TR SE and in section 8.1 of this SE.

Based on the above NRC staff evaluation, including the application of an Appendix B QA program, the endorsement of ASME.III.5 in RG 1.87, Rev. 3, and inclusion of the FUCH in the R&D program, the staff finds that the preliminary design of the FUCH meets the requirements of PDC 1.

Design Basis for Protection against Natural Phenomena (PDC 2)

The FUCH provides support for graphite and fuel pebbles during normal and seismic loading conditions and maintains the geometry of the fuel. PSAR table 3.7-2 indicates the FUCH is classified as Seismic Category I and, therefore, as described in PSAR Section 3.7.3, it is designed to withstand seismic loads associated with the safe-shutdown earthquake (SSE) without losing the capability to perform its required safety functions. PSAR section 6.4.5 states that the location of the FUCH within the SST and RXST protects it against the effects of natural phenomena other than seismic events (e.g., the effects of tornadoes, hurricanes, floods, tsunami, and seiches). The staff finds that Seismic Category I is an acceptable seismic design category for the FUCH because the FUCH is required to remain functional for safe shutdown of the reactor and, in accordance with Appendix S to 10 CFR Part 50, it is designed to withstand the effects of the SSE and remain functional per PSAR section 3.7. The staff finds the FUCH is protected from the effect of tornadoes, hurricanes, floods, tsunami, and seiches because it is located within the SST and is, therefore, protected from the effects of these phenomena by the SST.

Fire Protection (PDC 3)

PSAR section 6.4.5.1 states that the FUCH is located within the helium pressure boundary (HPB) and is constructed of metallic materials that are inherently non-combustible and non-fire sustaining. Based on the information provided in the PSAR section 6.4.8, the staff finds the preliminary design meets the requirements of PDC 3 because the FUCH is a passive component made of non-combustible materials and is contained within the HPB.

Environmental and dynamic effects design bases (PDC 4)

Staff reviewed PSAR Section 6.4.5, in which LME states that PDC 4 requirements on the design to accommodate the effects of and to be compatible with the environmental condition are satisfied by performing the environmental compatibility assessment and developing the material

qualification plan to mitigate the applicable material degradation modes, as described in PSAR Section 3.8. An assessment of irradiation effects on mechanical properties, an assessment of potential (de)carburization specific to Alloy 800H, and an assessment of erosion due to high-velocity helium flow is included in the R&D program as evaluated in section 13.3 of this SE. Staff also reviewed PSAR section 6.4.5 to assess whether it meets PDC 4 requirements of protecting the FUCH against dynamic effects. PSAR section 6.4.5 identified potential sources of dynamic hazards to the FUCH and stated that as the design matures, the credibility of these dynamic hazards will be systematically assessed and addressed. The staff will review the FSAR to confirm compliance of the FUCH design with PDC 4 regarding sufficient protection of FUCH from dynamic effects at the OL application stage.

Based on the evaluation above, the applicant has identified the FUCH as a major component that must meet PDC 4, and LME's planned assessment of material environmental compatibility in the applicant's R&D program and design development to support evaluation of credible dynamic effects provide reasonable assurance that compliance with PDC 4 will be satisfactorily resolved by the OL application stage in accordance with 10 CFR 50.35(a).

Sharing of structures, systems, and components (PDC 5)

PSAR section 6.4.5.1 states that each reactor module has its own dedicated FUCH. The staff reviewed arrangement drawings and confirmed that the functions of the FUCH are not shared between modules. Therefore, the staff finds that the FUCH meets the requirements of PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR Section 6.4.5, which states that periodic inspections of the FUCH are performed in accordance with the RIM program, which is evaluated in SE Section 8.2. The staff reviewed the potential MANDE requirements, including SSC health monitoring through temperature and radiation measurements, and helium-side chemistry monitoring. In addition, LME states that the preoperational testing, as described in PSAR Section 12.1.1, will be performed to provide a baseline for the inservice inspections. In sections 8.2 and 12.1.1 of this SE, the NRC staff finds that the RIM and preoperational testing program provide adequate assurance to maintain the functional capability of the safety-significant SSCs, including the FUCH. The NRC staff finds that the preliminary information about the FUCH RIM program meets the requirements of PDC 6. Considering the planned R&D addressing environmental compatibility, the high temperature operating conditions, and the importance of the FUCH's required functions, the staff will assess the adequacy of health monitoring and chemistry control as the only MANDE requirements at the OL application stage.

Reactor Design (PDC 10)

The staff reviewed the conformance of the FUCH design with PDC 10 for performing the functions of maintaining the overall core geometry to support core cooling and reactivity control and providing adequate heat removal for the fuel within the FUCH, in both cases to ensure the SARRDLs are not exceeded. The NRC staff notes that SARDDLs were not included in the CP application, but the NRC staff reviewed the PSAR dose consequence results for AOOs for a preliminary assessment, considering margin to the F-C target. PSAR section 6.4.5 states the FUCH provides acceptable design margins through (1) appropriate safety, quality and seismic classifications; (2) application of approved codes and standards; and (3) selection of materials

with consideration for the operating pressures and temperatures the FUCH would experience. The applicant also cites the comprehensive structural evaluation applied to the FUCH described in PSAR section 3.8, which includes a construction code assessment and environmental compatibility assessment. PSAR section 6.4.5.1 states that PDC 10 is a higher-level requirement that is satisfied by PDC 34 for the heat removal function, which the staff assesses in its evaluation of heat removal from the fuel within the FUCH. Based on the above assessment of the FUCH design, the staff finds the PSAR provides sufficient detail to describe how the FUCH meets the requirements of PDC 10.

Residual Heat Removal (PDC-34)

PDC RFDC 34 requires a passive means to effectively remove residual heat such that fuel and radionuclide release limits are not exceeded during DBEs and DBAs. PSAR section 5.3 aligns the design criterion and the associated safety function for components such as the FUCH by modifying the bases for PDC-RFDC 34 from the "Xe-100 Licensing Topical Report - Principal Design Criteria," to clarify that the intent of the phrase "effective heat removal" in PDC-RFDC 34 includes maintaining the integrity of the reactor core and the RPV, thereby ensuring the geometry required by PDC-RFDC 70 for passive heat removal. Similarly, PSAR section 5.3 clarifies the scope of PDC 70 by specifying that reactor systems capable of affecting core geometry, such as the FUCH, are designed such that the reactor core integrity is maintained and there is a low probability of rapidly propagating failure during DBEs and DBAs. Thus, PDC RFDC 34 addresses both the heat removal necessary to ensure fuel and radionuclide release limits are not exceeded for the fuel within the FUCH and the heat removal necessary to ensure the FUCH retains its capability to perform its function to maintain core geometry with a low probability of rapidly propagating failure. Because the FUCH is the first metallic component contacted by recently irradiated fuel exiting the core and the fuel pebbles are qualified to retain their function at higher temperatures, adequate heat removal to ensure the integrity of the FUCH is generally limiting.

PSAR section 6.4.5.1 states that the FUCH is designed to support passive and active heat transfer from the fuel within the FUCH. The RSF passive heat removal function involves transfer of heat by conduction, convection, and radiation from the fuel pebbles within the FUCH through the FUCH walls to the DCPV or RPV bottom head walls. The heat from the DCPV and RPV bottom head passes into the reactor cavity as described in PSAR Sections 6.4.8 and 6.4.6, respectively.

The applicant describes consideration of heat sources within the FUCH in PSAR section 6.4.5.1. The length of the FUCH is designed to provide sufficient decay heat reduction before the fuel exits the FUCH and enters the DCPV at a normal operating defueling rate. In addition, the applicant stated that an analysis would establish the length of time that fuel must remain within the reactor core before commencing a rapid full-core fuel offload, and the results of that analysis would be reflected in the FSAR as part of an OL application. PSAR section 6.4.5.1 also states that decay heat is the only credible source of heat generation within the FUCH after reactor shutdown because the geometry of the FUCH does not permit a pebble configuration more reactive than the fuel canister configurations that analyses show to remain safely subcritical under conservative conditions in PSAR section 3.9.

In PSAR section 6.4.5.1, the applicant also describes a preliminary thermal analysis of the FUCH. The applicant stated that the analysis used the GOTHIC model with conservative assumptions to compensate for uncertainties in the design. The applicant states the analysis demonstrated that the FUCH and DCPV peak temperatures during events involving a loss of

forced cooling would remain below code allowable limits. The applicant states that further evaluation will be performed with appropriate conservatisms, including the fuel pebble recirculation rate, and inclusion of the region of the DCPV that connects with the FHS that was not in the scope of the preliminary analysis. The evaluation will confirm the validity of the assumptions in the preliminary analysis and include appropriate modeling of heat removal pathways from the FUCH through the DCPV and ultimately to the reactor cavity cooling system (RCCS). The staff finds the scope and margin of the preliminary analysis are adequate, in accordance with 10 CFR 50.34(a)(4), to support conclusions regarding the preliminary design and to issue the CP. Detailed analyses would be required at the OL stage in accordance with 10 CFR 50.34(b)(4).

PDC CDC 34 requires an active means to transfer fission product decay heat and other residual heat from the reactor core to an ultimate heat sink at a rate such that SARRDLs are not exceeded during AOOs. The FUCH includes the following design feature to support active cooling: Holes in the single radial support for the FUCH and near the bottom of the FUCH allow helium flow downward from the RPV bottom head, between the DCPV and the FUCH, and up around the pebbles, absorbing heat along the way.

The staff evaluated the design bases of the FUCH against the criteria of PDC RFDC 34 and PDC CDC 34. The staff found that the FUCH preliminary design includes reasonable passive heat transfer paths and a means for active heat removal. The staff reviewed the preliminary thermal analysis and supporting documentation in audit and compared it with the preliminary configuration of the FUCH and surrounding SSCs depicted in PSAR Figure 6.4.5-1. The staff finds that the thermal analysis models the design of the FUCH, surrounding components, internal heat sources, and heat transport paths as the applicant describes in PSAR section 6.4.5.1. Also, as the applicant describes, the results of the preliminary analysis show substantial margin to code allowable temperatures for the selected FUCH material. Therefore, the staff finds the preliminary design bases for passive heat removal meets the requirements of PDC RFDC 34. In addition, PSAR section 6.4.5.1 describes a design feature of the FUCH that accommodates establishment and restoration of active primary flow cooling to the FUCH following AOOs, which the staff finds meets the requirements of PDC CDC 34.

Reactor Vessel and Reactor System Structural Design Basis (PDC-RFDC 70)

As evaluated for design basis conformance with PDCs 1, 2, and 6, the NRC staff determined the preliminary design of the FUCH has adequate structural integrity because the design and construction code is ASME Section III, Division 5, Subsection HG, as endorsed by RG 1.87, Rev. 3; it is designed to withstand the SSE and is protected from other natural phenomena; and the design includes provisions for suitable monitoring and inspection programs to provide assurance that degradation of the FUCH structural integrity would be detected. However, as discussed in relationship to PDC 4 above, the environmental compatibility of the materials has been identified for further R&D and design development to support evaluation and resolution of credible dynamic effects is ongoing. Thus, the FUCH provides sufficient structural support to maintain the core geometry for passive removal of the residual heat and sufficient insertion of the neutron absorbers, provided environmental compatibility and protection from credible dynamic effects is assured. Based on the evaluation above, the applicant has identified the FUCH as a major component that must meet PDC-RFDC 70, and LME's planned assessment of material environmental compatibility in the applicant's R&D program described in PSAR

section 13.3.2 and design development to support evaluation of credible dynamic effects provide reasonable assurance that compliance with PDC-RFDC 70 will be satisfactorily resolved by the OL application stage in accordance with 10 CFR 50.35(a).

Safety Functions

PSAR section 6.4.5.1 states that the FUCH supports performance of RSF 1.2.1 (Maintain core heat removal through passive means) and RSF 1.4.1 (Maintain core geometry). PSAR Table 5.3-1 identifies PDC-RFDC-34 (Residual Heat Removal) as the applicable design criterion for RSF 1.2.1 and PDC-RFDC 70 (Reactor structural design basis) as the applicable design criterion for RSF 1.4.1. PSAR Section 6.4.5.1 identifies the following SRDC for the FUCH:

- RSF 1.2.1.1 – The FUCH is designed to transfer sufficient heat via conduction, convection, or radiation to the DCPV and RPV to assure that fuel and radionuclide release limits are not exceeded during DBEs and DBAs. As the discussion for PDC 34 explains above, the FUCH material of construction is expected to be limiting with respect to establishing the sufficiency of heat removal.
- RSF 1.4.1.2 – The FUCH is designed to withstand stresses developed during DBEs and DBAs and ensure acceptable reactor core geometry is maintained to control reactivity and core heat removal.

As evaluated above regarding FUCH design basis conformance with PDC RFDC 34 and PDC RFDC 70, the staff finds that the preliminary design supports the required safety functions with appropriate criteria for reliability and capability defined. The staff also finds that the applicant describes appropriate plans for inclusion in a future OL application addressing: (1) further performance evaluation after the FUCH is added to the PRA logic model, operational conditions have been defined, and LBE challenges have been fully identified; and (2) research and development of the FUCH, as described in PSAR Section 13.3.2, to ensure code extension conditions and environmental challenges have been addressed.

PSAR Section 6.4.5.1 states that the FUCH also supports performance of PSF 1.2.2 (Maintain core heat removal through active means) and subsidiary PSF 1.2.2.2 (Restore active heat removal). The FUCH includes a design feature to support active cooling with cooled primary helium flow from the steam generators. This feature accommodates establishment and restoration of primary cooling flow to the FUCH following AOOs and provides assurance that SARRDLs are not exceeded.

The NRC staff finds the description of the FUCH safety functions to be consistent with the guidance for the content of CP applications found in section C.6.4, “Descriptions for SR SSCs,” of NEI 21-07, as endorsed by NRC RG 1.253, because the functions have been defined with preliminary performance requirements supported by plans for research and development and performance validation of the final design.

Technical Specifications

In Part IV of the CP application, “Technical Specifications,” the applicant identified the FUCH as a design feature technical specification based on its preliminary design to support cooling pebbles below 250°C prior to entering the fuel handling system (FHS). In addition, the applicant

states in PSAR section 6.4.5.1 that analyses reflected in the FSAR for the OL application will establish the duration that fuel pebbles must remain in the reactor core to ensure that the heat loads imposed on the FUCH (and other SSCs such as the FHS and fuel canisters) remain within their evaluated design limits during fuel transfer and storage.

Based on the review of the information provided in the CP application, as discussed above, the NRC staff finds the information provided meets 10 CFR 50.34(a)(5). The NRC staff finding is based on the variables, conditions, or items that are likely to become subjects of technical specifications for the FUCH being identified. Consistent with 10 CFR 50.34(b)(6)(vi), the NRC staff will evaluate the proposed technical specifications when detailed design information is available at the OL application stage.

Research and Development

10 CFR 50.34(a)(8) requires identification of any facility structures, systems, or components that necessitate research and development to confirm the adequacy of their design.

Table 13.3.2-1 of PSAR Chapter 13 identifies the SR FUCH within the scope of the proposed LME R&D program to resolve the safety question associated with the performance of its Alloy 800H material in the Xe-100 environmental operating conditions. The acceptability of inclusion of the SR metallics of the FUCH in the R&D program is discussed in Chapter 13 of this SE.

6.4.5.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 6.4.5 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter. Specifically, the information provided:

- meets requirements of 10 CFR 50.34(a)(3)(i) for the FUCH by describing the preliminary design identifying the applicable PDCs.
- meets the requirements of 10 CFR 50.34(a)(3)(ii) by describing how the preliminary FUCH design satisfies the applicable PDCs, except for PDC 4 and PDC-RFDC 70 that are being addressed through research and development affecting the FUCH.
- meets the requirements of 10 CFR 50.35(a) regarding incomplete qualification of the FUCH for environmental conditions under PDC 4 and PDC-RFDC 70 that is reasonably expected to be resolved through research and development and addressed in the operating license application.
- meets 10 CFR 50.34(a)(3)(iii) by providing information relative to materials of construction, general arrangement, and approximate dimensions in the FUCH, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.
- meets 10 CFR 50.34(a)(4) by describing preliminary and planned analyses and the safety margin for safety functions provided by the FUCH design.
- meets 10 CFR 50.34(a)(5) by identifying key design features of the FUCH as probable subjects of technical specifications.

- meets 10 CFR 50.34(a)(8) by identifying the research and development needs for the FUCH to confirm the performance of their metallic materials in the Xe-100 environmental operating conditions.
- meets relevant portions of 10 CFR 50.34(a)(12) by identifying that the FUCH is designed to retain its function for the safe shutdown earthquake.

6.4.6 Reactor Pressure Vessel

PSAR section 6.4.6 describes the reactor pressure vessel (RPV). The RPV supports the Control Rod Drive Mechanisms (CRDMs) and CMS. The RPV is part of the HPB and contains the CMS, GCA, and the pebbles. The RPV consists of vessel shell, top and bottom heads, closure flange and bolting, cross vessel nozzle, CRDM housings, nozzles for fuel inlet and outlet, instrumentation flange, and inspection/assembly person-way flanges and covers. The vessel shell is fabricated out of several ring forgings with welds that are located away from the "beltline" of the reactor to avoid the area of the highest neutron flux. The vessel shell has a single nozzle that acts as a flange to the cross vessel (CV). The CV nozzle will be either welded to the vessel shell or integrally forged. The RPV top head is a flat head with a bolted connection to the vessel flange, which is welded to the top of the vessel shell. The RPV top head consists of a vessel lid, a fuel reactor inlet pipe (FRIP), CRDM nozzle, and instrumentation flange. The vessel lid provides access to graphite blocks in the core structure and is attached to the top head by twenty studs. The FRIP provides a pathway for the fuel pebbles to enter the reactor through the top head. The FRIP enters the head at an angle to reduce the speed of the entering fuel. The FRIP is welded to the head with a full penetration weld at both the top and bottom of the top head. The CRDM nozzle and instrumentation flange provides connections for CRDM housings and instrumentation. The CRDM housings are threaded into the RPV top head. The RPV bottom head is shaped as a standard two-to-one semi-elliptical head. The head includes a nozzle for the fuel outlet piping and a person-way. The bottom head also includes connections for supports for the CMS.

System specific regulatory requirements applicable to the RPV are as follows:

- 10 CFR 50.34(a)(5)

System specific guidance applicable for the evaluation of the RPV is as follows:

- RG 1.87, Rev. 3, "Acceptability of ASME Code Section III, Division 5, 'High Temperature Reactors'," (ML25176A084).

In addition to PDCs 1 through 6, the following system specific PDCs are applicable to the RPV:

- PDC 10 - Reactor Design
- PDC-CDC 13 - Instrumentation and Control
- PDC 14 - Reactor Helium Pressure Boundary
- PDC-CDC 15 - Reactor Helium Pressure Boundary Design
- PDC-CDC 16 - Functional Containment Design
- PDC-RFDC 26 and PDC-CDC 26- Reactivity Control System
- PDC-CDC 30 - Integrity of Reactor Helium Pressure Boundary

- PDC 31 - Fracture Prevention of Reactor Helium Pressure Boundary
- PDC-RFDC 34 and PDC-CDC 34 - Residual Heat Removal
- PDC-RFDC 70 - Reactor Vessel and Reactor System Structural Design Basis

6.4.6.1 Technical Evaluation

Safety Significance

The RPV system is classified as SR based on its function to 1) withstand stresses and maintain geometry to ensure that the core and fuel geometry remains within design parameters (RSF 1.4.1.2); 2) provide pathways for control and shutdown rods to be inserted with eighteen control rod penetrations in the RPV top head (RSF 1.1.2.1); and 3) provide structural support for the core internal component geometries and tolerances needed to ensure adequate passive heat transfer from the core to the reactor cavity (RSF 1.2.1.2). The RSFs 1.1.2.1 and 1.2.1.2 that the RPV system performs to prevent or mitigate specific LBEs are modeled in the PRA, with the results summarized in Table 5.4-7. Per PSAR Section 5.4, RSF 1.4.1.2 is not explicitly included in the PRA logical model. However, RSF 1.4.1.2 is both preventive and mitigative in nature in that it prevents a larger release from occurring by reducing the frequency and consequence during events that challenge the structural integrity of the RPV. Based on the safety functions they perform, all the RPV components are classified as SR except CRDM housing, which is categorized as NSRST.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR section 6.4.6, which states that all the RPV safety significant components are designed to ASME BPVC Section III, Division 5, Subsection HB, Subpart A, which is endorsed by the NRC in RG 1.87, Rev. 3. Considering the design temperature of 285°C for the RPV components as shown in PSAR table 6.4.6-1, the NRC staff finds that Subsection HBA is the appropriate subsection for construction of the RPV because it is intended for Class A metallic pressure boundary components not exceeding the temperature limits in Table HAA-1130-1. Based on the information above, the NRC staff finds that the preliminary design of the RPV system meets the requirements of PDC 1 by using and supplementing appropriate Codes and Standards commensurate with the safety significance of the component safety functions. In addition, the staff found applying ASME BPVC Section III, Division 5 as a preliminary special treatment for the NSRST RPV components, such as the CRDM housings, to be acceptable, consistent with RG 1.87, Appendix A.

The full scope of the LMGS QAPD, described in PSAR section 8.1, is applied to the RPV. This program is designed to meet 10 CFR Part 50, Appendix B, and is appropriate for meeting the quality assurance requirements under PDC 1 for the SR aspects of the RPV. The NRC staff's evaluation of the LMGS QAPD is provided in the QAPD TR SE and in section 8.1 of this SE.

Based on the information above, the NRC staff finds that the preliminary design of the RPV meets the requirements of PDC 1.

Design Basis for Protection against Natural Phenomena (PDC 2)

The NRC staff reviewed PSAR section 6.4.6, which states that the RPV is designed to withstand seismic events and achieve performance levels corresponding to its designated seismic category as shown in PSAR table 3.7-2 and as discussed in PSAR section 3.7. As stated in PSAR table 3.7-2, the SR portions of the RPV are categorized as seismic Category I to ensure they can perform the associated RSFs during and following a seismic event. The NSRST component of the RPV (i.e., CRDM housing) is categorized as seismic Category IIA to ensure they can perform the associated NSRST PSFs during and following a seismic event. In addition, SSCs that could prevent the RPV SSCs from performing their RSF due to spatial interaction (i.e., seismic II/I) are categorized as seismic Category IIB to prevent this interaction. The seismic design requirements associated with each category and the codes and standards applied are described in PSAR section 3.7 and evaluated in section 3.7 of this SE. The NRC staff finds this seismic categorization and the associated design approach to be sufficient to address seismic aspects of PDC 2 for the preliminary design stage. Additional information on the final seismic design and the SSCs identified as needing seismic category IIB categorization to prevent interaction with this system will be reviewed at the OL application stage as part of the final design. PSAR section 6.4.6 also states that the RPV components are protected against natural phenomena, including tornadoes, external flooding, and extreme climate conditions as described in PSAR section 3.7, by their location within the SST and RXST. To evaluate the effects of flooding, the staff considered the DBHL information described in PSAR Chapter 2 and PSAR section 6.1. The staff review determined that the lowest elevation of the RPV is significantly higher than the site's still water flood elevation and, therefore, any potential water ingress into the reactor building would not pose a flooding hazard for the RPV.

Based on the evaluation above, the NRC staff finds that the RPV design meets the requirement of PDC 2.

Fire Protection (PDC 3)

The NRC staff reviewed PSAR section 6.4.6, which states that the probability that the RPV system is exposed to fires and explosions is minimized because 1) The RPV is made of high-temperature-resistant metallic materials that are inherently noncombustible and non-fire sustaining, and 2) The RPV and its interfacing structures are located in the reactor citadel where fixed combustible materials are negligible and transient combustibles are administratively controlled. In addition, PSAR section 6.4.6 states that the RPV system is protected by a comprehensive plant fire protection plan, which is described in PSAR section 8.6 and developed and maintained to meet the requirements of 10 CFR 50.48(a)(1) and PDC 3. The NRC staff evaluation of the LME fire protection plan is presented in section 8.6 of this SE.

Based on the above information, the NRC staff finds that the preliminary design of the RPV system design meets the requirements of PDC 3.

Environmental and dynamic effects design bases (PDC 4)

The NRC staff reviewed PSAR section 6.4.6, which states that PDC 4 requirements on the design to accommodate the effects of and to be compatible with the environmental conditions are satisfied by performing the environmental compatibility assessment and developing the material qualification plan to mitigate the applicable material degradation modes identified from the material compatibility assessment, as described in PSAR section 3.8. The NRC staff evaluated the environmental compatibility assessment method in section 3.8 of this SE and found that it meets the requirements of PDC 4.

PSAR section 6.4.6 refers to PSAR section 6.1.1 regarding the methods for the evaluation of dynamic effects from internal and external hazards to demonstrate that the preliminary design of the RPV system satisfies the requirements of PDC 4. As discussed in section 6.1 of this SE, the staff finds that LME's approach is appropriate to ensure SR SSCs will be adequately protected against the dynamic effect, including high energy line break and wind generated missiles. PSAR section 6.1.1.3 states that turbine missiles, missiles from rotating equipment and secondary missiles will be addressed during the OL application stage as part of the internal hazards assessment using PRA. The NRC staff will review LME's evaluation of turbine and other internally generated missiles and their impacts at the OL application stage, based on the final plant layout and design information.

Based on the above discussion, the NRC staff finds that the preliminary design of the RPV system meets the requirements of PDC 4. The NRC staff will evaluate the maturation of the materials qualification plans well as the final design during the OL application review. The staff will also confirm that the final design of the RPV is protected from internally generated missiles at the OL application stage based on the final plant layout and design information.

Sharing of structures, systems, and components (PDC 5)

PSAR section 6.4.6 states that the RPV components are not shared between reactor modules. It states that RPV components share a foundation system with multiple safety-significant SSCs. The adherence to PDC 5 within the context of a shared foundation system is described in PSAR section 3.7.6.3, and the NRC staff's finding that it is acceptable is discussed in section 3.7 of this SE. Based on the discussion above, the NRC finds that the preliminary design of the RPV system meets the requirements of PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR section 6.4.6, which states that periodic inspections of the RPV are performed in accordance with the RIM program, which is evaluated in section 8.2 of this SE. The RPV design provides access for implementing remote preservice and inservice requirements. The staff reviewed the potential MANDE requirements, including volumetric examination of structural welds, visual examination, SSC health monitoring through temperature and radiation measurements, and helium-side chemistry monitoring and found them appropriate on the basis that these methods are capable of identifying the principal degradation mechanisms expected for the RPV materials and operating environment, thereby providing adequate assurance to maintain the functional capability of the RPV. In addition, LME states that the preoperational testing, as described in PSAR section 12.1.1, will be performed to provide a baseline for the inservice inspections. Based on the discussion above, the NRC finds that the preliminary design of the RPV meets the requirements of PDC 6, and that the review of the RIM program can be reasonably left to a future OL application.

Reactor Design (PDC 10)

The NRC staff reviewed PSAR section 6.4.6, which states that PDC 10 is a higher-level requirement that is met in the RPV system design via application of PDC-CDC 26 and PDC-CDC 34. The NRC staff finds that this statement is appropriate because it is consistent with the

discussion in the NRC-approved Xe-100 PDC Topical Report. As discussed in the basis for PDC 10 in the Xe-100 PDC Topical Report, PDC 10 is a design limit that supports acceptance criteria in PDC-CDC 26 and PDC-CDC 34. PDC-CDC 26 and PDC-CDC 34 describe the requirements for the function of reactivity control and active heat removal to meet the SARRDL, respectively. The NRC staff notes that SARRDLs were not included in the CP application, but the NRC staff reviewed the PSAR dose consequence results for AOOs for a preliminary assessment, considering margin to the F-C target, and determined PDC 10 requirements are satisfied by meeting the requirements in PDC-CDC 26 and PDC-CDC 34. The NRC staff's evaluation of the RPV design's compliance with PDC-CDC 26 and PDC-CDC 34 is performed later in this SE section.

Instrumentation and Control (PDC-CDC 13)

The NRC staff reviewed PSAR section 6.4.6, which states that the RPV top head design includes an instrumentation flange that represents the interface with reactor core sensing devices and their supporting I&C systems. Therefore, the NRC staff finds that the preliminary design of the RPV system supports the instrumentation system to monitor variables and systems over their anticipated ranges for normal operation and during AOOs, thus meeting the requirements of PDC-CDC 13.

Reactor Helium Pressure boundary (PDC 14)

As discussed earlier in this section of the SE, the preliminary design of the RPV system complies with PDCs 1, 2, 4, and 6 to ensure the structural integrity of the RPV system. In addition, per PSAR section 6.4.6, ring forgings will be used to eliminate seam welds in the core region of the RPV. Seal welds are required at bolted interfaces between RPV components to prevent helium leakage, and seal welds are designed and constructed in compliance with ASME BPVC Section III, Division 5, which is endorsed by the NRC in RG 1.87, Revision 3.

Based on the above information, the NRC staff finds that the preliminary design of the RPV meets the requirements of PDC 14.

Reactor Helium Pressure Boundary Design (PDC-CDC 15)

As discussed earlier in this section of the SE, the preliminary design of the RPV system complies with PDCs 1, 2, 4, and 6 to ensure structural integrity of the RPV system. Therefore, the NRC staff finds that these design features are sufficient to ensure appropriate margins in the structural integrity of the RPV systems and thus satisfy the requirements of PDC-CDC 15.

Functional Containment Design (PDC-CDC 16)

As discussed earlier in this section of the SE, the preliminary design of the RPV system complies with PDCs 1, 2, 4, and 6 to ensure structural integrity of the RPV system. Therefore, the NRC staff finds that the preliminary design of the RPV system provides a barrier as part of the reactor functional containment to control the release of radioactivity to the environment, satisfying the requirements of PDC-CDC 16.

Reactivity Control System (PDC-RFDC 26 and PDC-CDC 26)

The NRC staff reviewed PSAR section 6.4.10, which states that LMGS's reactivity control and shutdown system (RCSS) consists of two separate rod banks, the reactivity control system (RCS) control rods and the reserve shutdown system (RSS) shutdown rods, which performs the reactivity control functions required in PDC-CDC 26 and PDC-RFDC 26, respectively. PSAR section 6.4.6 states that the RPV system includes eighteen CRDM housings that interface with the eighteen control rod penetrations in the RPV top head. The RPV top head penetrations align with the penetrations on the CB top plate and GCA sleeved channels to provide paths for movement of the nine RCS control rods and nine RSS shutdown rods used to control reactivity, as described in PSAR section 6.4.10.

Based on the above information, the NRC staff finds that the preliminary design of the RPV meets the requirements of PDC-RFDC and PDC-CDC 26 because it supports the insertion of the RCS control rods and RSS shutdown rods.

Integrity of Reactor Helium Pressure Boundary (PDC-CDC 30)

The NRC staff reviewed PSAR section 6.4.6, which states that the leakage detection requirements of PDC-CDC 30 are met largely via the appropriate application of PDC-CDC 13. In addition, the other sensors located throughout portions of the HPB primary helium flow path and their associated instrumentation are designed to ensure that excessive helium leakage, whether from the RPV or other portions of the HPB, is detected and (as practicable) isolated, as required in PDC-CDC 30.

Based on the above information, the NRC staff finds that the preliminary design of the RPV system meets the requirements of PDC-CDC 30.

Fracture prevention of reactor helium pressure boundary (PDC 31)

PSAR Section 6.4.6 refers to the methodology presented in PSAR Section 3.8 to evaluate the structural integrity of the safety significant metallic SSCs, including RPV, to demonstrate compliance with the requirements of PDC 31. As evaluated in section 3.8 of this SE, the NRC staff found that the structural integrity evaluation method and material qualification plan in PSAR section 3.8 appropriately considers the service temperature, service degradation of material properties, and failure modes of the SSCs under all the applicable operating conditions in AOOs, DBEs, and DBAs, as well as the uncertainties in material properties and the effect of service conditions on material properties as required by PDC 31. The staff finds the use of ASME BPVC Section III, Division 5, as assessed in SE section 3.8, will ensure that appropriate service level conditions and bounding load combinations are accounted for in the design. In addition, ASME BPVC Section III, Division 5 rules support PDC 31 by accounting for residual stresses and requiring construction weld examination, which reduces the likelihood of unacceptable flaws from going into service. Therefore, the NRC staff finds that the preliminary design of the RPV system meets the requirements of PDC 31.

Residual Heat Removal (PDC-RFDC 34 and PDC-CDC 34)

The NRC staff reviewed PSAR section 6.4.6, which states that in accordance with PDC-RFDC 34, the passive heat removal function is achieved via the heat transfer path from the core,

through the GCA and CMS, and on to the RPV wall, the reactor cavity, the RCCS, and finally to the environment. The RPV design provides structural support for the core internal component geometries and tolerances needed to ensure adequate passive heat transfer. In addition, PSAR section 6.4.6 states that the RPV maintains the core internal structure and features that direct helium flow through and around the core, thus supporting the active heat transfer requirements in PDC-CDC 34. The staff's current assessment of PDC-RFDC 34 and PDC-CDC 34 is based on the preliminary thermal-hydraulic evaluations performed as part of the LBE consequence analysis. These preliminary analyses included the RPV within the core model and are evaluated in sections 3.1.2 and 3.6 for this SE. The NRC staff will evaluate the adequacy of passive and active heat transfer for the final design and analysis as part of the thermal-hydraulic review during the OLA.

Based on the above information, the NRC staff finds that the preliminary design of the RPV meets the requirements of PDC-RFDC and PDC-CDC 34.

Reactor Structural Design Basis (PDC-RFDC 70)

The NRC staff reviewed PSAR section 6.4.6, which states that the RPV provides support to the CMS, GCA, and fuel by providing support to the base pedestal of the CMS, which in turn supports the GCA and fuel pebble bed. PSAR section 6.4.3 states that **[[** [redacted] **]]** PSAR section 6.4.6 states that the RPV top head maintains the upper boundary of the core geometry through its connection to the RPV flange.

As discussed in the staff's evaluation pertaining to compliance with PDCs 1, 2, 4, 6, and PDC-RFDCs 26 and 34, the preliminary design of the RPV system has adequate structural integrity and thus provides sufficient support to maintain the core geometry for passive removal of the residual heat and sufficient insertion of the neutron absorbers.

Based on the above information, the NRC staff finds that the preliminary design of the RPV system meets the requirements of PDC-RFDC 70.

Safety Functions

PSAR Section 6.4.6 states that the RPV is designed to support RSFs 1.1.2.1 (Provide means to insert negative reactivity to ensure safe shutdown), 1.2.1.2 (transfer heat from vessel to reactor cavity), and 1.4.1.2 (limit RPV stress to acceptable levels). LMGS identified PDC-RFDCs 26, 34, and 70 as applicable design criteria for the RPV to support RSFs 1.1.2.1, 1.2.1.2, and 1.4.1.2.

PSAR section 6.4.6 states that the RPV supports NSRST PSFs 1.1.3.1 (control core reactivity to preclude reactor trips) and 1.2.2.2 (restore active heat removal), as well as PSFs 2.7.1, 2.7.2, and 2.7.3 (Control the frequency of depressurization events). LMGS identified PDC-CDCs 13, 15, 16, 26, 30, and 34 to support these PSFs. The NRC staff finds that these PDC selections align with the RSFs and NSRST PSFs identified for the preliminary design of RPV because they are consistent with PSAR tables 5.3-1 and 5.3-2, which map PDCs with RSFs and PSFs based on PRA results

Based on the evaluation above, the NRC staff finds that the preliminary design of the RPV provides adequate support for RSFs 1.1.2.1, 1.2.1.2, and 1.4.1.2, and PSFs 1.1.3.1, 1.2.2.2, and 2.7.1 through 2.7.3.

Technical Specifications

Part IV of the CP application identified the RPV to have technical specifications based on the preliminary design and the associated preliminary safety analysis, with additional details provided in Part IV, "Technical Specifications." The proposed preliminary SLs include an upper bound for the RPV pressure and temperature to maintain core geometry (RSF 1.4.1) and control heat removal (RSF 1.2.1), respectively. LME states that the analytical limits to ensure that a SL is not exceeded will be provided in the TS for a future OL application. The preliminary proposed design features include that 1) the RPV controls heat removal by passively transferring heat from the fuel, through the CMS and GCA, to the reactor cavity, RCCS, and environment, and 2) the RPV is designed to maintain core geometry by limiting RPV stress to acceptable levels. Table IV-3 lists RPV temperature as an LCO variable to support RSFs 1.2.1 and 1.4.1.

Based on the review of the information provided in the CP application, as discussed above, the NRC staff finds the information provided meets the requirements of 10 CFR 50.34(a)(5). The staff's finding is based on the variables, conditions, or items that are likely to become subjects of technical specifications for the facility being identified in Part IV of the CP application. Consistent with 10 CFR 50.34(b)(6)(vi), the NRC staff will evaluate the proposed technical specifications when detailed design information is available at the OL application stage.

6.4.6.2 Conclusion

The staff concludes that the preliminary RPV is acceptable and meets the requirements of 10 CFR 50.34(a)(3)(i), 10 CFR 50.34(a)(3)(ii), 10 CFR 50.34(a)(3)(iii), and 10 CFR 50.34(a)(5). This conclusion is based on the following:

- Based on the information provided in PSAR sections 6.4.6.1, the applicant has met 10 CFR 50.34(a)(3)(i) for the RPV by identifying PDCs applicable to the preliminary design of the RPV and describing the preliminary design of the RPV.
- Based on the information provided in PSAR sections 6.4.6.1 and 6.4.6.2, the applicant has met 10 CFR 50.34(a)(3)(ii) by describing how the preliminary design of RPV meets the PDCs identified.
- Based on the information provided in PSAR sections 6.4.6.1 and 6.4.6.2, the applicant has met 10 CFR 50.34(a)(3)(iii) by providing information relative to materials of construction, general arrangement, and approximate dimensions in the RPV, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.
- Based on the information provided in Part IV to the CP, the applicant has met 10 CFR 50.34(a)(5) by identifying probable subjects of technical specifications for the RPV.

6.4.7 Reactor Vessel Support System

PSAR section 6.4.7 describes the reactor vessel support system (RVSS). The RVSS consists of bottom support beams attached to the reactor bottom head and a seismic ring support, which is mounted to the bottom support and surrounds the RPV near the RPV top head.

System specific guidance applicable for the evaluation of the RVSS is as follows:

- Interim Staff Guidance (ISG) DANU-ISG-2023-01 "Material Compatibility for non-Light Water Reactors," (ML22203A175).
- RG 1.26, Rev. 5, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants Staff Regulatory Guidance 1," (ML16082A501).
- RG 1.29, Revision 6, "Seismic Design Classification for Nuclear Power Plants," (ML21155A003).

In addition to PDCs 1 through 6, the following system specific PDCs are applicable to the RVSS:

- PDC-RFDC 70 – Reactor structural design basis
- PDC-RFDC 71 – Reactor support structure building design basis

6.4.7.1 Technical Evaluation

Safety Significance

The RVSS is classified as SR based on its function to maintain the position of the RPV within the reactor compartment such that the reactor system geometry and integrity are maintained. Specifically, the RVSS supports performance of RSF 1.4.2, "Maintain reactor support geometry," as stated in PSAR section 6.4.7.1. By performing this function, the RVSS prevents breach of the RPV and loss of core geometry during LBEs that challenge the structural support of the units (e.g., design basis seismic event). These events are not currently modeled within the PRA, but the function performed is both preventive and mitigative in nature in that it prevents a larger release from occurring by reducing the frequency and consequence during events that challenge the structural support of the units.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR section 6.4.7.1, which states that the RVSS is designed to the 2023 Edition of ASME BPVC Section III, Division 5, Subsection HF, which is endorsed by the NRC staff in RG 1.87, Rev. 3. Subsection HF is appropriate for construction of the RVSS because it is intended for Class A and Class B metallic core supports not exceeding the temperature limits in Table HAA-1130-1. The full scope of the LMGS QAPD, described in PSAR section 8.1, is applied to the RVSS. This program is designed to meet 10 CFR Part 50, Appendix B, and is appropriate for meeting the quality assurance requirements under PDC 1 for the SR aspects of the RVSS. The NRC staff's evaluation of the LMGS QAPD is provided in the QAPD TR SE and in section 8.1 of this SE

Based on the information above, the NRC staff finds that the preliminary design of the RVSS meets the requirements of PDC 1.

Design Basis for Protection against Natural Phenomena (PDC 2)

The NRC staff reviewed PSAR section 6.4.7, which states that the RVSS is designed to withstand seismic events and achieve performance levels corresponding to its designated seismic category as shown in PSAR Table 3.7-2 and as discussed in PSAR section 3.7. As stated in PSAR table 3.7-2, the RVSS is categorized as seismic Category I to ensure they can perform the associated RSFs during and following a seismic event. In addition, SSCs that could prevent the RPV SSCs from performing their RSF due to spatial interaction (i.e., seismic II/I) are categorized as seismic Category IIB to prevent this interaction. The seismic design requirements associated with each category and the codes and standards applied are described in PSAR section 3.7 and evaluated in section 3.7 of this SE. The NRC staff finds this seismic categorization and the associated design approach to be sufficient to address seismic aspects of PDC 2 for the preliminary design stage. Additional information on the final seismic design and the SSCs identified as needing seismic category IIB categorization to prevent interaction with this system will be reviewed at the OL application stage as part of the final design. PSAR section 6.4.7 also states that the RVSS components are protected against natural phenomena, including tornadoes, external flooding, and extreme climate conditions as described in PSAR Section 3.7, by their location within the SST. To evaluate the effects of flooding the staff considered the DBHL information described in PSAR Chapter 2 and section 6.1. The staff's review determined that the lowest elevation of the RVSS is higher than the site's still water flood elevation, and therefore the potential water ingress into the structure where the RVSS is located would not pose a flooding hazard for the RVSS.

Based on the evaluation above, the NRC staff finds that the RVSS design meets the requirement of PDC 2.

Fire Protection (PDC 3)

The NRC staff reviewed PSAR section 6.4.7, which states that the probability that the RVSS system is exposed to fires and explosions is minimized because 1) The RVSS is made of high-temperature-resistant metallic materials that are inherently noncombustible and non-fire sustaining, and 2) The RVSS and its interfacing structures are located in the reactor citadel where fixed combustible materials are negligible and transient combustibles are administratively controlled. In addition, PSAR Section 6.4.7 states that the RVSS system is protected by a comprehensive plant fire protection plan, which is described in PSAR Section 8.6 and developed and maintained to meet the requirements of 10 CFR 50.48(a)(1) and PDC 3.

Based on the above information, the NRC staff finds that the preliminary design of the RVSS system design meets the requirements of PDC 3.

Environmental and dynamic effects design bases (PDC 4)

The NRC staff reviewed PSAR section 6.4.7, which states that PDC 4 requirements on the design to accommodate the effects of and to be compatible with the environmental conditions are satisfied by performing the environmental compatibility assessment and developing the

material qualification plan to mitigate the applicable material degradation modes identified from the material compatibility assessment, as described in PSAR section 3.8. The NRC staff evaluated the environmental compatibility assessment method in section 3.8 of this SE and found that it meets the requirements of PDC 4.

PSAR section 6.4.7 refers to PSAR Section 6.1.1 to address the methods for the evaluation of dynamic effects from internal and external hazards to demonstrate that the preliminary design of the RVSS system satisfies the requirements of PDC 4. Per section 6.1 of this SE, the staff found that LME's approach is appropriate to ensure SR SSCs will be adequately protected against the dynamic effect, including high energy line break and wind generated missiles. PSAR section 6.1.1.3 states that turbine missiles, missiles from rotating equipment and secondary missiles will be addressed during the OL application stage as part of the internal hazards assessment using PRA. The NRC staff will review LME's evaluation of turbine and other internally generated missiles and their impacts at the OL application stage, based on the final plant layout and design information.

Based on the above discussion, the NRC staff finds that the preliminary design of the RVSS system meets the requirements of PDC 4. Staff will review LME's evaluation of turbine and other internally generated missiles and their impacts at the OL application stage, based on the final plant layout and design information.

Sharing of structures, systems, and components (PDC 5)

The NRC staff reviewed PSAR section 6.4.7, which states that the RVSS components are not shared between reactor modules. It also states that RVSS components share a foundation system with multiple safety-significant SSCs. The adherence to PDC 5 within the context of a shared foundation system is described in PSAR Section 3.7.6.3 and the NRC staff's finding that it is acceptable is discussed in section 3.7 of this SE. Based on the discussion above, the NRC finds that the preliminary design of the RVSS system meets the requirements of PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR section 6.4.7, which states that periodic inspections of the RVSS components are performed in accordance with the RIM program, which is evaluated in SE section 8.2. The RVSS design provides access for implementing remote preservice and inservice requirements. The NRC staff reviewed the identified potential MANDE requirements, including visual examination and SSC health monitoring through temperature and radiation measurements, and finds them appropriate on the basis that these examinations and monitoring activities provide sufficient information to ensure timely detection of degradation affecting RVSS functional capability. The NRC staff finds that the preliminary design of the RVSS meets the requirements of PDC 6, with review of the RIM program reasonably being left to the OL application.

Reactor Structural Design (PDC-RFDC 70) and Reactor Support Structure Design Basis (PDC-RFDC 71)

The NRC staff reviewed PSAR section 6.4.7 and finds that the RVSS design aligns with PDC-RFDCs 70 and 71 through supporting the RPV, CMS, GCA and fuel pebbles, transferring loads to the reactor cavity floor during all the LBEs, and maintaining the RPV position to ensure sufficient insertion of the neutron absorbers. Based on the evaluation for PDCs 1, 2, 4, and 6, the NRC staff determined the RVSS components are designed to withstand stresses to perform its function of maintaining core geometry for all the operating conditions.

Based on the above information, the NRC staff finds that the preliminary design of the RVSS meets the requirements of PDC-RFDCs 70 and 71.

Safety Functions

PSAR Section 6.4.7 states that the RVSS is designed to support RSF 1.4.2 (maintain reactor support geometry). LMGS identified PDC-RFDC 71 (reactor support structure design basis) as an applicable design criterion for the RVSS to support RSFs 1.4.2. In addition, LMGS identified PDCs 1 through 6, with which the RVSS design complies. The NRC staff finds that these PDC selections align with the RSFs and NSRST PSFs identified for the preliminary design of RVSS because they are consistent with PSAR tables 5.3-1 and 5.3-2, which map PDCs with RSFs and PSFs based on PRA results. The NRC staff's evaluation also finds that the preliminary design of the RVSS complies with these PDCs as evaluated above under Compliance with Applicable PDCs in this SE section.

Based on the evaluation above, the NRC staff finds that the preliminary design of RVSS provides adequate support for RSF 1.4.2.

6.4.7.2 Conclusion

The staff concludes that the preliminary RVSS is acceptable and meets the requirements of 10 CFR 50.34(a)(3)(i), 10 CFR 50.34(a)(3)(ii), and 10 CFR 50.34(a)(3)(iii). This conclusion is based on the following:

- Based on the information provided in PSAR section 6.4.7.1, the applicant has met 10 CFR 50.34(a)(3)(i) for the RVSS by identifying PDCs applicable to the preliminary design of the RVSS and describing the preliminary design of the RVSS.
- Based on the information provided in PSAR section 6.4.7.1, the applicant has met 10 CFR 50.34(a)(3)(ii) by describing how the preliminary design of RVSS meets the PDCs applicable to the preliminary design of the RVSS.
- Based on the information provided in PSAR section 6.4.7.1, the applicant has met 10 CFR 50.34(a)(3)(iii) by providing information relative to materials of construction, general arrangement, and approximate dimensions in the RVSS, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.

6.4.8 Defuel Chute Pressure Vessel

PSAR section 6.4.8 describes the defuel chute pressure vessel (DCPV). The DCPV consists of an upper cylindrical section that is attached to the reactor vessel bottom head and surrounds the lower portion of the FUCH and a T-shaped elbow section that attaches to the bottom of the cylindrical section, supports the lower portion of the FUCH, and receives the fuel pebbles exiting the FUCH. The DCPV forms part of the HPB and provides a path for cooled helium from the steam generators to flow around the FUCH, which holds fuel pebbles discharged from the core until they have cooled sufficiently to enter the DCPV. The DCPV upper cylinder connects to the RPV bottom head by a bolted flanged joint. The DCPV upper cylinder connects to the DCPV elbow at another bolted flanged connection. At the outlets of the DCPV, the elbow is bolted at each end to the flanged connections of the two sphere unloading machines (SUM), which are described as part of the FHS in PSAR section 7.3.6.

The DCPV design and service level conditions are listed in PSAR table 6.4.8-1, the DCPV is constructed from SA-508 Grade 3, Class 1, and the fasteners for all joints are SA-540, Grade B23, Class 3. The DCPV is ASME Code Class A and constructed in accordance with ASME Section III, Division 5, Subsection HB, with specified design basis temperatures indicating low temperature service rules apply (Subpart A).

System specific regulatory requirements applicable to DCPV are as follows:

- 10 CFR 50.34(a)(5), which requires identification of preliminary variables and conditions that are probable subjects of technical specifications.
- 10 CFR 50.34(a)(12), which requires compliance with the earthquake engineering criteria in Appendix S to 10 CFR Part 50.

System specific guidance for the evaluation of the DCPV is as follows:

- Regulatory Guide 1.87, Revision 3, “Acceptability of ASME Code Section III, Division 5, 'High Temperature Reactors,'” (ML25176A084).

- Interim Staff Guidance (ISG) DANU-ISG-2023-01 “Material Compatibility for non-Light Water Reactors,” (ML22203A175).

In addition to PDCs 1 through 6, the following PDCs are applicable to the DCPV:

- PDC 10 - Reactor Design
- PDC-RFDC 13 and PDC-CDC 13 - Instrumentation and Control
- PDC 14 - Reactor Helium Pressure Boundary
- PDC-CDC 15 - Reactor Helium Pressure Boundary Design
- PDC-CDC 16 - Functional Containment Design
- PDC-CDC 30 - Integrity of Reactor Helium Pressure Boundary
- PDC 31 - Fracture Prevention of Reactor Helium Pressure Boundary
- PDC-RFDC 34 and PDC-CDC 34 - Residual Heat Removal

- PDC-RFDC 70 - Reactor Vessel and Reactor System Structural Design Basis

6.4.8.1 Technical Evaluation

Safety Significance

The DCPV serves as the helium pressure boundary (HPB) from the bottom head of the RPV to the fuel handling system (FHS) and encases the FUCH. The upper cylindrical portion of the DCPV is attached by a flanged connection to the RPV and provides a helium coolant flow path in the annular region between the DCPV and the FUCH. The vertical entry to the T-shaped elbow is attached by a flanged connection to the bottom of the DCPV upper cylinder and the horizontal portion of the elbow is attached to the two sphere unloading machines (SUMs) (one at each end) by flanged connections. The T-shaped elbow portion of the DCPV receives the fuel pebbles exiting the bottom of the FUCH and provides structural support to the lower portion of the FUCH.

The DCPV has a required safety function to maintain core geometry, which it accomplishes by supporting the FUCH and holding the fuel spheres in the core region (RSF 1.4.1). As PSAR section 5.4 notes, the DCPV performance of the maintain core geometry function has not been explicitly included in the PRA logic model prepared for the CP application. However, as part of the HPB, the DCPV has a PRA safety function to control the frequency of reactor depressurization events (PSF 2.7). This related function of the DCPV is included in the PRA logic model and is risk-significant because it contributes to maintaining or restoring active core cooling (PSF 1.2.2.2) and retention of radionuclides (PSF 2). These functions are accomplished by adequate design and monitoring of the condition of the DCPV during normal operations and accident conditions.

The DCPV supports heat removal from the FUCH and contained fuel pebbles. The DCPV upper cylinder provides passive radial heat transfer from the FUCH and the fuel contained within the FUCH to the reactor cavity area (RSF 1.2.1). The DCPV elbow collects fuel spheres after they pass through the FUCH, provides adequate cooling by transferring heat to the reactor building and limiting the heat generation within the DCPV to decay heat by maintaining the fuel spheres in a subcritical geometry (RSF 1.2.1), and directs the spheres to the SUMs for entry into the FHS. The upper cylindrical portion of the DCPV provides a path for cooled helium returning from the steam generators to move downward around the outside of the FUCH when the HPB is pressurized and the helium circulators are in operation (PSF 1.2.2.2). These heat removal functions are relied upon primarily to maintain the DCPV and FUCH material conditions within code allowable values so that the function to maintain core geometry is not challenged.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR section 6.4.8, which states that the DCPV is constructed in accordance with ASME BPVC Section III, Division 5, Subsection HBA, which is endorsed by the NRC staff in RG 1.87, Rev. 3. Considering the design temperature of 285°C for the DCPV as

shown in PSAR table 6.4.8-1, ASME BPVC Section III, Division 5, Subsection HBA is an appropriate Subsection because it is intended for Class A metallic pressure boundary components not exceeding the temperature limits in Table HAA-1130-1. Based on the information above, the NRC staff finds that the preliminary design of the DCPV system meets the requirements of PDC 1 regarding use of appropriate Codes and Standards, supplemented or modified as necessary to assure a quality product commensurate with the safety significance of the component safety functions.

The full scope of the LMGS QAPD, described in PSAR section 8.1, is applied to the DCPV. This program is designed to meet 10 CFR Part 50, Appendix B, and is appropriate for meeting the quality assurance requirements under PDC 1 for the SR aspects of the DCPV. The NRC staff's evaluation of the LMGS QAPD is provided in the QAPD TR SE and in section 8.1 of this SE. Based on the information above, the NRC staff finds that the preliminary design of the DCPV meets the requirements of PDC 1.

Design Basis for Protection against Natural Phenomena (PDC 2)

PSAR table 6.4.8-1 lists the DCPV Design and Service Level Conditions. PSAR table 3.7-2 indicates the DCPV is classified as Seismic Category I and, as described in PSAR section 3.7.3, it is designed to withstand seismic loads associated with the safe-shutdown earthquake (SSE) without losing the capability to perform its required safety functions. PSAR section 6.4.8.1 states that the location of the DCPV within the SST and RXST protects it against the effects of natural phenomena other than seismic events (e.g., the effects of tornadoes, hurricanes, floods, tsunami, and seiches).

The staff finds that Seismic Category I is an acceptable seismic design category for the DCPV because the DCPV is required to remain functional for safe shutdown of the reactor and, consistent with the Appendix S to 10 CFR Part 50, it is designed to withstand the effects of the SSE and remain functional per PSAR section 3.7. PSAR section 6.4.1.2 states that the SST is designed to withstand tornado/hurricane missile impacts and that SST openings prevent water intrusion by being located above flood height, designed to be substantially impermeable, or designed with an equivalent preventative measure. Additionally, the staff review determined that the DCPV is at an elevation that is significantly higher than the site's still water flood elevation. Therefore, the staff finds the DCPV is protected from the effects of tornadoes, hurricanes, floods, tsunami, and seiches because it is located within the SST and protected from the effects of these phenomena by the SST. Based on the review performed, the staff finds the preliminary design of the DCPV is consistent with PDC 2 because it is protected from these phenomena by the SST and designed to withstand earthquakes without loss of capability to perform its safety functions.

Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize, consistent with other safety requirements and the safety significance of the functions to be performed, the probability and effect of fires and explosions. PSAR section 6.4.8.1 states that the DCPV is constructed from metallic materials that are inherently non-combustible and is in the reactor citadel below the RPV, where there is negligible fire hazard. Based on the information provided in PSAR section 6.4.8, the staff finds the preliminary design is consistent with PDC 3 because

the DCPV is a passive component made of non-combustible materials in an area with a very low fire hazard.

Environmental and dynamic effects design bases (PDC 4)

The NRC staff reviewed PSAR section 6.4.8.1, which states that PDC 4 requirements on the design of the DCPV to accommodate the effects of and to be compatible with the environmental conditions are satisfied by performing the environmental compatibility assessment and developing the material qualification plan to mitigate the applicable material degradation modes identified from the material compatibility assessment, as reviewed in section 3.8 of this SE.

The staff also reviewed PSAR section 6.4.8.1 to assess whether the facility design meets PDC 4 requirements for protection of the DCPV against dynamic effects. PSAR section 6.4.8.1 identified potential sources of dynamic hazards to the DCPV and stated that as the design matures, the credibility of these dynamic hazards will be systematically assessed and addressed. The staff will review the FSAR to confirm compliance of the facility design with PDC 4 regarding sufficient protection of the DCPV from dynamic effects during the OL application review.

Based on the above discussion, the NRC staff finds that the preliminary design of the DCPV system meets the requirements of PDC 4.

Sharing of structures, systems, and components (PDC 5)

PSAR section 6.4.8.1 states that each module has its own dedicated DCPV. The staff finds the description sufficient to determine the DCPV is not a shared SSC between multiple nuclear power units and therefore meets the requirements of PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

PSAR section 6.4.8 states that periodic inspections of the DCPV are performed in accordance with the RIM program, which is discussed in SE section 8.2, and that the DCPV design provides access for implementing remote preservice and inservice requirements. The applicant identified potential MANDE requirements, including visual examination, SSC health monitoring through temperature and radiation measurements, and helium-side chemistry monitoring in concert with the chemistry program described in PSAR section 8.8.

The NRC staff evaluated the described monitoring and inspection capability. Because the DCPV design provides access and proposed monitoring and inspection addresses credible DCPV degradation mechanisms, the staff finds that the preliminary monitoring measures provide reasonable assurance that the functional capability of the DCPV will be maintained. Therefore, the NRC finds that the preliminary design of the DCPV meets the requirements of PDC 6 with respect to permitting necessary access for the proposed inspection and monitoring, with review of the final RIM program implementation reasonably being left to the OL application.

Reactor Design (PDC 10)

The staff reviewed the conformance of the DCPV design with PDC 10 to assess: (1) maintaining the overall core geometry to support core cooling and reactivity control to ensure the SARRDLs

are not exceeded; and (2) providing adequate heat removal for the fuel within the FUCH to ensure the SARRDLs are not exceeded. PSAR section 6.4.8.1 states that the DCPV provides acceptable design margins through (1) appropriate safety, quality and seismic classifications; (2) application of approved codes and standards; and (3) selection of materials with consideration for the operating pressures and temperatures the DCPV components would experience. The NRC staff notes that parameters and values for SARDDLs were not identified in the CP application, but NRC staff reviewed the PSAR preliminary dose consequence results for AOOs and finds acceptable margins to the F-C target. The applicant also cites the comprehensive structural evaluation applied to the DCPV described in PSAR section 3.8, which includes a construction code assessment and environmental compatibility assessment. PSAR section 6.4.8.1 states that PDC 10 is a higher-level requirement that is satisfied by PDC 34 for the heat removal function, which the staff will employ in its evaluation of heat removal from the fuel within the FUCH. Based on the above assessment of the DCPV design, the staff finds the PSAR provides sufficient detail to describe how the DCPV meets the requirements of PDC 10.

Instrumentation and Control (PDC 13)

The NRC staff reviewed PSAR section 6.4.8.1, which states that sensors located in portions of the HPB other than within the DCPV and the associated instrumentation are designed to ensure that excessive helium leakage is detected and isolated. These sensors will ensure that all depressurizations and thermal transients within the design basis are detected and appropriate plant responses, including reactor and circulator trips, occur in a timely fashion to prevent the DCPV and FUCH from exceeding their design temperature. The applicant states that these capabilities will be informed by PRA evaluations of break sizes and frequencies and transient safety analyses that will be available for the OL application.

The staff evaluated the information provided in PSAR section 6.4.8.1 related to instrumentation and control. The staff recognizes that continued circulator operation without active heat removal in the steam generators, with or without reactor trip, could challenge the DCPV and lower RPV head by circulating hot helium potentially capable of heating those components above their design temperatures. However, the PSAR indicates both the RPS and the IPS independently stop circulator operation when necessary for plant protection, suggesting the function has very high reliability. PSAR section 5.2 identifies the capability to detect and isolate HPB failures leading to depressurizations as safety significant to support radionuclide retention. The staff finds that the applicant has described suitable instrumentation and controls for the preliminary design stage to monitor systems, provide indication of the function of the DCPV, and control variables, such as helium temperature and pressure, within prescribed operating ranges during AOO and DBE conditions, which meets the requirements of PDC 13..

Reactor Helium Pressure Boundary (PDC 14), Reactor Helium Pressure Boundary Design (PDC-CDC 15), Functional Containment Design (PDC-CDC 16), and Fracture Prevention of Reactor Helium Pressure Boundary (PDC 31)

These design criteria are evaluated together because they are addressed by related elements of the DCPV design.

The NRC staff reviewed PSAR section 6.4.8.1, which states that these PDC are satisfied by appropriate classification of the DCPV as SR, use of the approved ASME BPVC Section III, Division 5, Subsection HBA codes, and appropriate material selection in combination with the comprehensive metallic structural evaluation described in PSAR section 3.8. These actions serve to minimize the potential for design conditions of the DCPV to be exceeded or for material degradation that results in leakage because the structural evaluation addressed in PSAR

section 3.8 includes consideration of service conditions, failure mechanisms, and environmental compatibility. Therefore, the NRC staff finds that, by using DCPV design and materials conforming to accepted codes and by incorporating consideration of service conditions, failure mechanisms, and environmental compatibility, the DCPV design basis reasonably addresses the design criteria of PDC14, PDC-CDC 15, PDC-CDC16, and PDC 31.

Integrity of Reactor Helium Pressure Boundary (PDC-CDC 30)

The staff reviewed PSAR section 6.4.8.1, which states that the features that support identification of HPB leakage are the same as the sensors and instrumentation that monitor key plant process parameters to detect and, to the extent practicable, isolate HPB leakage as described in the evaluation of PDC-CDC 13 above. The NRC staff finds, in accordance with PDC-CDC 30, that the reactor design includes appropriate sensors and instrumentation to identify HPB leakage, as described in the staff's evaluation of PDC-CDC 13 for the DCPV. In addition, the DCPV acts as a continuation of the RPV to contain and support the FUCH, which precludes an HPB isolation function within the DCPV. Instead, the SUMs, which are attached to the DCPV, include the HPB isolation function. Because the DCPV constitutes part of the HPB that cannot practically be isolated, the location of the isolation capability within the SUMs, as evaluated in Section 7.3.6 of this SE, is acceptable.

Residual Heat Removal (PDC-RFDC 34)

PSAR section 5.3 aligns the design criterion and the associated safety function for components such as the DCPV by modifying the bases for PDC-RFDC 34 from the Xe-100 PDC TR to clarify that the intent of the phrase "effective heat removal" in PDC-RFDC 34 includes maintaining the integrity of the reactor core and the RPV, thereby ensuring the geometry required by PDC-RFDC 70 for passive heat removal. Similarly, PSAR section 5.3 clarifies the scope of PDC 70 by specifying that reactor systems capable of affecting core geometry, such as the DCPV, are designed such that the reactor core integrity is maintained and there is a low probability of rapidly propagating failure during DBEs and DBAs. Thus, PDC-RFDC 34 addresses both the heat removal necessary to ensure fuel and radionuclide release limits are not exceeded for the fuel within the DCPV and the heat removal necessary to ensure the DCPV retains the capability to perform its function to maintain core geometry with a low probability of rapidly propagating failure. Because the DCPV is designed for low temperature service in accordance with ASME BPVC Section III, Division 5, Subsection HBA, and the fuel pebbles are qualified to retain their function at higher temperatures, adequate heat removal to ensure the integrity of the DCPV is generally limiting.

The NRC staff reviewed PSAR section 6.4.8.1, which states that the DCPV is designed to support passive and active heat transfer from the fuel within the FUCH. The RSF passive heat removal function involves transfer of heat by conduction, convection, and radiation from the fuel pebbles within the lower portion of the FUCH through the FUCH walls to the DCPV. The heat from the DCPV passes into the reactor cavity as described in PSAR section 6.4.8.

The applicant describes consideration of heat sources within the DCPV in PSAR section 6.4.8.1. The length of the FUCH is designed to provide sufficient decay heat reduction before the fuel enters the DCPV at a normal operating defueling rate. PSAR section 6.4.8.1 states that decay heat is the only credible source of heat generation within the DCPV because the geometry of the DCPV elbow region does not permit a pebble configuration more reactive than the fuel canister configurations that analyses show to remain safely subcritical under conservative conditions in PSAR section 3.9.

In PSAR section 6.4.8.1, the applicant also describes a preliminary thermal analysis of the FUCH. The applicant stated that the analysis used the GOTHIC model with conservative assumptions to compensate for uncertainties in the design. The applicant states the analysis demonstrated that the FUCH and DCPV peak temperatures during events involving a loss of forced cooling would remain below code allowable limits. The applicant states that further evaluation will be performed with appropriate conservatisms, including the pebble recirculation rate and the region of the DCPV that connects with the FHS that was not in the scope of the preliminary analysis. The evaluation will confirm the validity of the assumptions in the preliminary analysis and include appropriate modeling of heat removal pathways from the FUCH through the DCPV and ultimately to the reactor cavity cooling system (RCCS). The DCPV supports active cooling of the fuel by providing a flow path for cooled helium from the steam generator around the FUCH.

The NRC staff evaluated the design bases of the DCPV against the criteria of PDC-RFDC 34 and PDC-CDC 34. The staff finds that the DCPV preliminary design includes reasonable passive heat transfer paths and a means for active heat removal. The NRC staff reviewed the preliminary thermal analysis and supporting documentation in audit and compared it with the preliminary configuration of the DCPV, FUCH, and surrounding SSCs depicted in PSAR figure 6.4.5-1. The NRC staff finds that the thermal analysis models the design of the DCPV upper cylinder, nearby components, internal heat sources, and heat transport paths as the applicant describes in PSAR section 6.4.8.1. Also, as the applicant describes, the NRC staff agrees that the results of the preliminary analysis show substantial margin to code allowable temperatures for the selected DCPV material. For the analysis of the final design, the applicant stated that the DCPV elbow region will be included in a full scope evaluation considering changes in fuel recirculation rate and using thermal models validated for the moderately complex configuration of the DCPV. Based on the completed preliminary analysis of the DCPV cylinder, the margin available to temperature limits, and the relevance of the analysis to the DCPV elbow, the NRC staff finds the preliminary design bases for passive heat removal meets the requirement of PDC-RFDC 34. In addition, PSAR section 6.4.8.1 describes, and PSAR figure 6.4.5-1 shows, a design feature of the DCPV that accommodates establishment and restoration of active primary flow around the FUCH following AOOs, which the staff finds meets the requirement of PDC-CDC 34.

Reactor Vessel and Reactor System Structural Design Basis (PDC-RFDC 70)

The DCPV addresses the requirement of PDC-RFDC 70 by confining the pebbles exiting the FUCH and providing structural support to hold up the column of pebbles. As evaluated for design basis conformance with PDCs 1, 2, 4, 6, 10, 14, 15, 16, and 31, the preliminary design of the DCPV has adequate structural integrity and thus provides sufficient support to maintain the core geometry for passive removal of the residual heat and sufficient insertion of the neutron absorbers. The NRC staff finds the preliminary DCPV meets the requirement of PDC-RFDC 70 because the design and construction code is ASME Section III, Division 5, Subsection HBA, as endorsed by RG 1.87, Rev. 3; the DCPV design is subject to the criteria for design of metallic SR SSCs outlined in PSAR section 3.8; and periodic assessment of conditions during operation will be performed in accordance with the RIM program, as described in PSAR section 8.2, to address environmental conditions that may be challenging to the DCPV.

Safety Functions

PSAR section 6.4.8.1 states that the DCPV supports performance of RSF 1.2.1 (Maintain core heat removal through passive means) and RSF 1.4.1 (Maintain core geometry). PSAR table 5.3-1 identifies PDC-RFDC-34 as the applicable design criterion for RSF 1.2.1 and PDC-RFDC 70 as the applicable design criterion for RSF 1.4.1. PSAR Section 6.4.8.1 identifies the following SRDC for the DCPV:

- RSF 1.2.1.1 – The DCPV is designed to ensure that sufficient heat is transferred via radiation and convection to the reactor cavity to assure that fuel and radionuclide release limits are not exceeded during DBEs and DBAs. As the discussion for PDC-RFDC 34 explains above, the DCPV material of construction is expected to be limiting with respect to establishing the sufficiency of heat removal.
- RSF 1.4.1.2 – The DCPV is designed to withstand stresses developed during DBEs and DBAs and ensure acceptable reactor core geometry is maintained to control reactivity and core heat removal.

As evaluated above regarding the DCPV design basis conformance with PDC-RFDC 34 and PDC-RFDC 70, the staff finds that the preliminary design supports the required safety functions with appropriate criteria for reliability and capability defined. The staff also finds that the applicant describes appropriate plans for inclusion in a future OL application of further performance evaluation after the DCPV is added to the PRA logic model, operational conditions have been defined, and LBE challenges have been fully identified.

PSAR section 6.4.8.1 states that the DCPV supports performance of PSF 1.2.2 (Maintain core heat removal through active means) and subsidiary PSF 1.2.2.2 (Restore active heat removal). The DCPV design includes a feature to support active cooling with cooled primary helium flow from the steam generators. As evaluated above regarding DCPV design basis conformance with PDC-CDC 34, the staff finds that this feature accommodates establishment and restoration of primary cooling flow to the DCPV and FUCH following AOOs and provides assurance that SARRDLs are not exceeded.

PSAR section 6.4.8.1 also states that the DCPV supports performance of PSF 2 (Retain radionuclides in the HPB) and PSF 2.7 (Maintain HPB pressure integrity). As evaluated above regarding DCPV design basis conformance with PDC-CDC 13, PDC 14, PDC-CDC 15, PDC-CDC 16, PDC-CDC 30, and PDC 31, the staff finds that the design of the DCPV provides assurance that the integrity of the HPB will be maintained with suitable margin to satisfy PSF 2 and PSF 2.7.

The NRC staff finds the description of the DCPV safety functions to be consistent with the guidance for the content of CP applications found in section C.6.4, “Descriptions for SR SSCs,” of NEI 21-07, as endorsed by NRC RG 1.253, because the functions have been defined with preliminary performance requirements and are supported by plans for further development and performance validation of the final design.

Technical Specifications

The applicant identified certain variables and conditions for technical specifications for the DCPV in Part IV of the CP application. The proposed limiting conditions for operation identified in Table IV-3 of the proposed technical specifications (TS) relate to reactor building

temperature, reactor vessel temperature, and FHS isolation valve status. The basis provided in Part IV of the CP application describes retaining the pressure integrity of the DCPV and the DCPV function of retaining radionuclides, which align with PSF 2.7 and PSF 2, respectively.

The staff reviewed information provided in the PSAR related to the proposed TS parameters and conditions. The proposed parameters of reactor building temperature and reactor vessel temperature provide indication that initial conditions for the required function of passive cooling for the DCPV are satisfied, which helps to ensure that the code allowable temperatures for the DCPV would not be exceeded following DBEs and DBAs. The condition of the FHS isolation valves helps to ensure HPB breaks can be promptly isolated to minimize potential for the break to adversely affect the DCPV integrity through excessive high temperature helium flow or dynamic effects from the break. Since the DCPV evaluation in the PRA logical model was limited to small pressure boundary failures, additional evaluation of licensing basis events is necessary, as the applicant described in PSAR section 6.4.5.1 for changes in fuel recirculation rate and in PSAR section 6.4.8 regarding high temperature helium circulation, to establish if additional parameters or conditions are necessary for TS at the OL application stage. The NRC staff finds the information provided meets the requirements of 10 CFR 50.34(a)(5) because reasonable parameters and conditions that are likely to become subjects of TS related to preliminary analysis for the facility are identified. The NRC staff will evaluate the content of the TS for compliance with 10 CFR 50.34(b)(6)(vi) at the OL application stage when detailed design and safety analysis information for the DCPV and related SSCs is available.

6.4.8.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 6.4.8 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP. Specifically, the information provided:

- meets requirements of 10 CFR 50.34(a)(3)(i) for the DCPV by describing the preliminary design and identifying the applicable PDCs.
- meets the requirements of 10 CFR 50.34(a)(3)(ii) by describing how the DCPV design satisfies the applicable PDCs.
- meets 10 CFR 50.34(a)(3)(iii) by providing information relative to materials of construction, general arrangement, and approximate dimensions in the DCPV, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.
- meets 10 CFR 50.34(a)(4) by describing preliminary and planned analyses and the safety margin for safety functions provided by the design of the DCPV.
- meets 10 CFR 50.34(a)(5) by identifying preliminary FHS variables and conditions that are probable subjects of technical specifications.
- meets relevant portions of 10 CFR 50.34(a)(12) by identifying that the DCPV is designed to retain its function for the safe shutdown earthquake.

6.4.8.3 References

- Interim Staff Guidance (ISG) DANU-ISG-2023-01 “Material Compatibility for non-Light Water Reactors,” (ML22203A175).

6.4.9 Primary Pressure Relief System (PPRS)

PSAR section 6.4.9 describes the primary pressure relief system (PPRS). The PPRS provides overpressure protection for the HPB. The PPRS is a combination of safety relief valves (SRVs), three-way valves, a power operated relief valve (PORV), and rupture discs connected by piping. The rupture disc connections are above the elevation of the expected water level following a steam generator tube rupture (SGTR). The PPRS connects to the steam generator pressure vessel (SGPV) and provides overpressure protection to ensure the integrity of the HPB. The PPRS design considers the collective volumes of the HPB and any LBEs that could result in an overpressurization to ensure that sufficient pressure relief capacity is provided to maintain core geometry within acceptable limits.

System specific regulatory requirements applicable to the PPRS are as follows:

- 10 CFR 50.34(a)(5)

System specific guidance applicable for the evaluation of the PPRS is as follows:

- Interim Staff Guidance (ISG) DANU-ISG-2023-01, “Material Compatibility for non-Light Water Reactors,” (ML22203A175).
- RG 1.26, Rev. 5, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants Staff Regulatory Guidance 1,” (ML16082A501).
- RG 1.29, Revision 6, “Seismic Design Classification for Nuclear Power Plants,” (ML21155A003).

In addition to PDCs 1 through 6, the following system specific PDCs are applicable to the PPRS:

- PDC 10 - Reactor Design
- PDC-CDC 13 - Instrumentation and Control
- PDC 14 - Reactor Helium Pressure Boundary
- PDC-CDC 15 - Reactor Helium Pressure Boundary Design
- PDC-CDC 16 - Functional Containment Design
- PDC-CDC 30 - Integrity of Reactor Helium Pressure Boundary
- PDC 31 - Fracture Prevention of Reactor Helium Pressure Boundary

- PDC-RFDC 70 - Reactor Structural Design Basis

6.4.9.1 Technical Evaluation

Safety Significance

The PPRS system is classified as safety related (SR) based on its function to provide sufficient pressure relief capacity to maintain core geometry within acceptable limits during DBEs and DBAs. The PPRS system supports performance of RSF 1.4.1.1, "Control primary pressure." The safety functions that the PPRS system performs to prevent or mitigate specific LBEs are modeled in the PRA, with the results summarized in PSAR table 5.4-11. As stated in PSAR section 6.4.9, the relief valves and rupture discs in the PPRS are SR components that support RSF 1.4.1.1, while the three-way valves and piping are NSRST SSCs that support PSF 2.7, "Maintain HPB pressure integrity."

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR tables 6.4.9-1 and 6.4.9-2, which show that the SR and NSRST components of the PPRS are designed to the 2023 Edition of ASME BPVC Section III, Division 5, Subsection HB, Subpart A and to the 2022 Edition of ASME B31.1, respectively. The 2023 Edition of ASME BPVC Section III, Division 5 is endorsed by the NRC staff in RG 1.87, Rev. 3, as an appropriate approach for the design and construction of the SR components in nuclear power plants. Considering the design temperature of 370°C for the SR components as shown in PSAR table 6.4.9-1, Subsection HB, Subpart A is the appropriate subsection for construction of the SR components in the PPRS because it is intended for Class A metallic pressure boundary components not exceeding the temperature limits in Table HAA-1130-1. For the NSRST components in the PPRS, PSAR section 6.4.9 states that the non-nuclear code ASME B31.1 adequately addresses all structural failure modes with appropriate margins for the NSRST commensurate with their safety significance and the low design temperature of 370°C. PSAR section 6.4.9 also states that the potential need for enhancements to ASME B31.1 is being evaluated because the PPRS is part of the HPB. LME identified the need for special treatments of enhanced weld integrity measures that are more stringent than baseline B31.1 requirements, such as requiring 100% volumetric examination (UT or RT) of pressure boundary fusion welds and requiring full penetration boundary fusion welds. The NRC staff finds the identified special treatments are appropriate to demonstrate adequacy of the selected codes and standards per the requirements of PDC 1 as the enhanced weld integrity measures would support HPB integrity.

The full scope of the LMGS QAPD, described in PSAR section 8.1, is applied to the SR PPRS SSCs. This program is designed to meet 10 CFR Part 50, Appendix B, and is appropriate for meeting the quality assurance requirements under PDC 1 for the SR aspects of the SR PPRS SSCs. The NSRST PPRS SSCs are designed, fabricated, constructed, and tested in accordance with the quality assurance program described in Part III of the LMGS QAPD and PSAR section 8.1. Additional special treatments related to quality assurance will be specified at the OL application stage. Application of Part III of the LMGS QAPD at the CP stage is sufficient as a preliminary special treatment to meet the quality assurance requirements under PDC 1 for the NSRST portions of the PPRS. This finding is based on the NRC staff's evaluation of the LMGS QAPD in the QAPD TR SE and in section 8.1 of this SE.

Based on the above information, the NRC staff finds that the preliminary design of the PPRS meets the requirements of PDC 1. The NRC staff will review the special treatments of enhanced weld integrity to supplement ASME B31.1 selected for the final design of the NSRST components during the OL application stage.

Design Basis for Protection against Natural Phenomena (PDC 2)

The NRC staff reviewed PSAR section 6.4.9, which states that the PPRS components are designed to withstand seismic events and achieve performance levels corresponding to their designated seismic category as shown in PSAR table 3.7-2 and as discussed in PSAR section 3.7. As stated in PSAR table 3.7-2, the SR portions of the PPRS are categorized as seismic Category I to ensure they can perform the associated RSFs during and following a seismic event. NSRST components of the PPRS are categorized as seismic Category IIA to ensure they can perform the associated NSRST PSFs during and following a seismic event. In addition, SSCs that could prevent the PPRS SSCs from performing their RSF due to spatial interaction (i.e., seismic II/I) are categorized as seismic Category IIB to prevent this interaction. The seismic design requirements associated with each category and the codes and standards applied are described in PSAR section 3.7 and evaluated in section 3.7 of this SE. The NRC staff finds this seismic categorization and the associated design approach to be sufficient to address seismic aspects of PDC 2 for the preliminary design stage. Additional information on the final seismic design and the SSCs identified as needing seismic category IIB categorization to prevent interaction with this system will be reviewed at the OL application stage as part of the final design. Per PSAR section 6.4.9, the PPRS components are located inside the SST and are protected from natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions as described in PSAR section 3.7. The NRC staff's evaluation of these items is in section 3.7 and 6.4.1 of the SE. To evaluate the effects of flooding, the NRC staff considered the DBHL information described in PSAR Chapter 2 and section 6.1. The NRC staff review determined that the lowest elevation of the PPRS is higher than the site's still water flood elevation and therefore the potential water ingress into the structure where the PPRS is located would not pose a flooding hazard for the PPRS.

Based on the above information, the NRC staff finds that the preliminary design of the PPRS meets the requirements of PDC 2.

Fire Protection (PDC 3)

PSAR section 6.4.9 states that the probability that the PPRS is exposed to fires and explosions is minimized because 1) The PPRS is made of high-temperature-resistant metallic materials that are inherently noncombustible and non-fire sustaining, and 2) The PPRS and its interfacing structures are located in the steam generator support structure area within the SST where fixed combustible materials are negligible and transient combustibles are administratively controlled. In addition, PSAR section 6.4.9 states that the PPRS is protected by a comprehensive plant fire protection plan, which is described in PSAR Section 8.6, and developed and maintained to meet the requirements of 10 CFR 50.48(a)(1) and PDC 3.

Based on the above information, the NRC staff finds that the preliminary design of the PPRS design meets the requirements of PDC 3.

Environmental and dynamic effects design bases (PDC 4)

The NRC staff reviewed PSAR section 6.4.9 on the compliance of the PPRS preliminary design with the requirements in PDC 4 to accommodate the effects of and to be compatible with the

environmental conditions and found that these requirements are satisfied by performing the environmental compatibility assessment and developing the material qualification plan to mitigate the applicable material degradation modes identified from the material compatibility assessment, as described in PSAR section 3.8. The NRC staff evaluated the environmental compatibility assessment method in section 3.8 of this SE and found that it meets the requirements of PDC 4.

For the requirements on protection against dynamic effects, PSAR section 6.4.9 states that as the design matures, the sources of dynamic hazards to the PPRS will be assessed systematically, and for dynamic hazards determined to be credible, analyses will be performed to demonstrate adequate PPRS protection. Design measures to mitigate credible dynamic effects include incorporation of component supports and shielding as well as strategic equipment location. LME states that the credibility assessment, final design approach, supporting analyses, and demonstration that dynamic effects have been appropriately addressed will be documented in the FSAR to be submitted with a future OL application. The NRC staff finds LME's approach to ensure adequate protection of the PPRS from dynamic effect acceptable because it provides a structured, commitment-backed process to evaluate dynamic hazards as the design matures.

Based on the above information, the NRC staff finds that the preliminary design of the PPRS design meets the requirements of PDC 4. Staff will evaluate the materials qualification program as well as the final design at the OL application stage.

Sharing of structures, systems, and components (PDC 5)

The NRC staff reviewed PSAR section 6.4.9, which states that the PPRS components are not shared between reactor modules. It also states that PPRS components share a foundation system with multiple safety-significant SSCs. The adherence to PDC 5 within the context of a shared foundation system is described in PSAR section 3.7.6.3, and the NRC staff found it acceptable in section 3.7 and 6.4.1 of this SE. Based on the discussion above, the NRC finds that the preliminary design of the PPRS system meets the requirements of PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR section 6.4.9 for compliance of the PPRS preliminary design to the requirements in PDC 6 for both SR and NSRST components.

For SR components, LME states that periodic inspections of the safety related passive PPRS components are performed in accordance with the RIM program, which is evaluated in SE section 8.2. The PPRS design provides access for implementing remote preservice and inservice requirements. The NRC staff reviewed the identified potential MANDE requirements, including visual examination of the SR valve bodies, online leakage detection, and helium side chemistry monitoring. Because the PPRS design provides access and proposed monitoring and inspection addresses credible degradation mechanisms, the staff finds that the preliminary monitoring measures provide reasonable assurance that the functional capability of the PPRS will be maintained. Functional testing of the SR PPRS active components will be performed in accordance with the IST program, which is evaluated in SE section 8.4. In addition, LME states that the preoperational testing as described in PSAR section 12.1.1 will be performed to provide a baseline for the inservice inspections, which is evaluated in SE section 12.1.1.

As confirmed by NRC staff via a Request for Confirmatory Information (ML26239A157), LME is in the process of incorporating the results of new risk-significance evaluations following the NEI 18-04 methodology. Based on a preliminary evaluation, LME identified the PPRS as risk-significant, potentially impacting the scoping of NSRST SSCs in operational programs. As stated in PSAR sections 8.2 and 8.4, SSCs that are risk significant will be included in the RIM and IST programs and staff will evaluate the operational program scoping of SSCs in the PPRS system at the OL application stage based on final risk-significance determinations.

For NSRST components, LME states that periodic inspections of the NSRST PPRS passive components are performed in accordance with the RAM program, which is discussed in SE section 8.10. The PPRS design provides access for implementing remote preservice and inservice requirements. Staff reviewed the identified potential MANDE requirements, including visual examination of the NSRST valve bodies, online leakage detection, and helium side chemistry monitoring and finds them appropriate to provide adequate assurance to maintain the functional capability of the NSRST PPRS SSCs. LME states that functional testing of the NSRST PPRS active components will be performed in accordance with the maintenance program, which is evaluated in SE section 8.3. The NRC staff reviewed potential IST requirements, including visual testing and replacement of rupture discs, as well as testing of valves, and finds them appropriate to provide adequate assurance to maintain the functional capability of the NSRST active SSCs. However, the maintenance program does not specify a method for developing and implementing an IST program, such as ASME OM-2. The NRC staff will review the final NSRST ISI and IST programs at the OL application stage. In addition, LME states that the preoperational testing, as described in PSAR section 12.1.1 and evaluated in SE section 12.1.1, will be performed to provide a baseline for the inservice inspections.

Based on the discussion above, the NRC staff finds that the preliminary design of the SR PPRS SSCs meets the requirements of PDC 6. The NRC staff will evaluate the RAM and maintenance programs as part of the OL application review, including the basis and acceptability of the departure from the scope of ASME Section XI-2 as endorsed in RG 1.246 (SE section 8.2) and OM-2 as endorsed in RG 1.220 (SE Section 8.4) which includes SSCs that perform functions requiring special treatment for defense in depth adequacy.

Reactor Design (PDC 10)

Per the NRC-approved Xe-100 PDC Topical Report, PDC-10 is a higher-level requirement that is met via application of PDC-CDC 26 and PDC-CDC 34. PDC-CDC 26 requires that the reactor be designed to provide movable poisons that can insert negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the SARRDL and HPB design limits are not exceeded, and safe shutdown is achieved and maintained during AOOs. PDC-CDC 34 requires the design of an active means to transfer fission product decay heat and other residual heat from the reactor core to an ultimate heat sink at a rate such that SARRDL limits are not exceeded during AOOs. The NRC staff notes that SARRDLs were not included in the CP application, but NRC staff reviewed the PSAR dose consequence results for AOOs for a preliminary assessment, considering margin to the F-C target.

The NRC staff finds that the preliminary design of the PPRS satisfies the requirements of PDC-CDC 26 and PDC-CDC 34 by controlling primary pressure to provide overpressure protection for the RPV system, which supports insertion of negative reactivity and active heat transfer. Therefore, staff finds that the preliminary design of PPRS satisfies the requirements of PDC 10.

Instrumentation and Control (PDC-CDC 13)

The NRC staff reviewed PSAR section 6.4.9, which states that the PPRS design includes interfaces with reactor system sensing devices and their supporting I&C systems. This design feature supports monitoring variables and systems (e.g., helium pressure, temperature, and helium leakage) over their anticipated ranges during normal operation and AOOs. Therefore, the NRC staff finds that the preliminary design of the PPRS is consistent with the requirements in PDC-CDC 13.

Reactor Helium Pressure boundary (PDC 14)

As discussed earlier in this section, the preliminary design of the PPRS complies with PDCs 1, 2, 4, and 6 to ensure the structural integrity of the PPRS, and thus ensures an extremely low probability of abnormal leakage and unacceptable moisture ingress. Therefore, the NRC staff finds that the preliminary design of the PPRS satisfies the requirements of PDC 14.

Reactor Helium Pressure Boundary Design (PDC-CDC 15)

As discussed earlier in this section, the preliminary design of the PPRS complies with PDCs 1, 2, 4, and 6 to ensure structural integrity of the PPRS. The NRC staff finds that these design features are sufficient to ensure appropriate margins in the structural integrity of the PPRS, as discussed above in the NRC staff's evaluation of those PDCs, and thus satisfies the requirements of PDC-CDC 15.

Functional Containment Design (PDC-CDC 16)

As discussed earlier in this section, the preliminary design of the PPRS complies with PDCs 1, 2, 4, and 6 to ensure structural integrity of the PPRS. In addition, the NRC staff noted that the rupture discs and pressure relief valves directly vent to the reactor building shield structure per PSAR section 6.4.9.1. The performance of the PPRS with respect to functional containment is included in the mechanistic source term (MST) modeling described in PSAR section 3.1.4 and evaluated in section 3.1.4 of this SE. MST analyses account for the impact of the PPRS in functional containment, either through it forming part of the pressure boundary in cases where the boundary remains intact and the pressure relief does not actuate, or in cases where the pressure relief actuates and the valves re-close to provide some level of additional radionuclide retention. However, as discussed in section 3.1.4 of this SE, the NRC staff notes that for specific events such as those resulting in failure of pressure relief valves to close or rupture disks burst into the shield structure, the CP consequence analyses take no credit for the holdup, decay, or deposition in the shield structure for the consequence analysis. The appropriate treatment of these impacts of the PPRS in the MST and the demonstration that the LBEs meet the F-C target and integrated risk metrics indicate that the functional containment design aspects of the PPRS are appropriately addressed. Therefore, the NRC staff finds that the preliminary design of the PPRS provides a sufficient barrier as part of the reactor functional containment to control the release of radioactivity to the environment, satisfying the requirements of PDC-CDC 16.

Integrity of Reactor Helium Pressure Boundary (PDC-CDC 30)

The NRC staff reviewed PSAR section 6.4.9, which states that the leakage detection requirements of PDC-CDC 30 are met largely via the appropriate application of PDC-CDC 13. In addition, the sensors located throughout other portions of the HPB primary helium flow path and their associated instrumentation are designed to ensure that excessive helium leakage, whether from the PPRS or other portions of the HPB, is detected and, as practicable, isolated, as required by PDC-CDC 30.

Based on the above information, the NRC staff finds that the preliminary design of the PPRS system meets the requirements of PDC-CDC 30.

Fracture prevention of reactor helium pressure boundary (PDC 31)

The NRC staff reviewed PSAR section 6.4.9, which refers to the methodology to evaluate the structural integrity of the safety significant metallic SSCs, including the PPRS, in PSAR section 3.8, to demonstrate compliance with the requirements of PDC 31 in the preliminary design of the PPRS. As evaluated in section 3.8 of this SE, the NRC staff finds that the structural integrity evaluation method and material qualification plan in PSAR section 3.8 appropriately considers the service temperature, service degradation of material properties, and failure modes of the SSCs under all the applicable operating conditions in AOOs, DBEs, and DBAs, as well as the uncertainties in material properties and the effect of service conditions on material properties as required in PDC 31. The staff finds the use of ASME BPVC Section III, Division 5 for the SR components, and ASME B31.1 for the NSRST components, will ensure that bounding load combinations are accounted for in the design of the SR components. In addition, ASME BPVC Section III, Division 5 rules for SR components and special treatments of enhanced weld integrity measures for NSRST components support PDC 31 by requiring construction weld examination, which reduces the likelihood of unacceptable flaws going into service. Therefore, based on the discussion above, the NRC staff finds that the preliminary design of the PPRS system meets the requirements of PDC 31.

Reactor Structural Design Basis (PDC-RFDC 70)

The NRC staff reviewed PSAR section 6.4.9, which states that the PPRS is designed to provide sufficient pressure relief capacity to maintain core geometry within acceptable limits during DBEs and DBAs. Per PSAR section 6.4.9.1, the PPRS design includes redundant pressure relief flow paths and pressure relief devices (e.g., SRVs and rupture discs), each sized to maintain HPB pressure below the maximum 110-percent design pressure limit specified in ASME BPVC Section III, Division 5. In addition, the description of the PPRS performance and operation in PSAR section 6.4.9.2 states the following:

1. The SRVs that automatically open when the pressure exceeds the valve set pressures are sized and rated based on the anticipated system responses to the most limiting SGTR with only one valve functioning.
2. The rupture disks will actuate if the pressure exceeds their setpoint which is lower than the SRV setpoint.
3. The PPRS mitigation response is the same for all LBEs involving overpressure conditions and the PPRS performs its RSF in response to environmental challenges modeled in LBEs and normal operations.

4. The PPRS performs its RSF with no reliance on operator action, electrical power, logic circuits or interlocks, controls and displays, or support systems.

Based on the above information, the NRC staff finds that the preliminary design of the PPRS provides sufficient pressure relief capacity to maintain core geometry within acceptable limits, thus satisfying the requirements of PDC-RFDC 70.

Safety Function

PSAR section 6.4.9 states that the PPRS is designed to support RSF 1.4.1, "Maintain core geometry." LMGS identified PDC-RFDC 70, "Reactor structural design basis," as the applicable design criterion for the PPRS to support RSF 1.4.1. PSAR section 6.4.9 also states that the PPRS supports NSRST PSFs 2.7.1, "Control the frequency of small depressurization events," 2.7.2, "Control the frequency of medium depressurization events," and 2.7.3. "Control the frequency of large depressurization events." LME identified PDC-CDCs 13, 15 16, and 30 to support the PSFs 2.7.1 through 2.7.3. The NRC staff finds that these PDC selections align with the RSF and NSRST PSFs identified for the preliminary design of PPRS because they are consistent with PSAR tables 5.3-1 and 5.3-2, which map PDCs with RSFs and PSFs based on PRA results The NRC staff finds that the PPRS design features satisfy the required PDCs as evaluated above in this SE section.

Technical Specifications

Part IV of the CP application identified PPRS to have technical specifications based on the preliminary design and the associated preliminary safety analysis. The proposed list of preliminary proposed technical specifications includes SRV setpoints based on HPB pressure and the number of SRVs available as LCOs to support RSF 1.4.1, "Maintain core geometry."

Based on the review of the information provided in the CP application, as discussed above, the NRC staff finds the information meets 10 CFR 50.34(a)(5). The staff's finding is based on the variables, conditions, or items that are likely to become subjects of technical specifications for the facility being identified in Part IV of the CP application. Consistent with 10 CFR 50.34(b)(6)(vi), the NRC staff will evaluate the proposed technical specifications when detailed design information is available at the OL application stage.

6.4.9.2 Conclusion

The NRC staff concludes that the preliminary PPRS design is acceptable. This conclusion is based on the following:

- Based on the information provided in PSAR sections 6.4.9.1, the applicant has met 10 CFR 50.34(a)(3)(i) for the PPRS by identifying PDCs applicable to the PPRS and describing the preliminary design of the PPRS.
- Based on the information provided in PSAR sections 6.4.9.1 and 6.4.9.2, the applicant has met 10 CFR 50.34(a)(3)(ii) by describing how the PPRS meets the PDCs identified.
- Based on the information provided in PSAR sections 6.4.9.1 and 6.4.9.2, the applicant has met 10 CFR 50.34(a)(3)(iii) by providing information relative to materials of construction, general arrangement, and approximate dimensions in the PPRS, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.

- Based on the information provided in Part IV to the CP, the applicant has met 10 CFR 50.34(a)(5) by identifying probable subjects of technical specifications for the PPRS.

6.4.10 Reactivity Control and Shutdown System

The NRC staff reviewed PSAR section 6.4.10, which describes the reactor control and shutdown system (RCSS). The RCSS system provides an independent and diverse means of reactor shutdown. The system is designed to provide means for maintaining safe shutdown conditions during LBEs. The RCSS consists of two separate rod banks:

- *Reserve shutdown system (RSS)*: consists of nine neutron absorber rods, fully withdrawn during normal operations and fully inserted to maintain shutdown conditions as needed.
- *Reactivity control system (RCS)*: consists of nine neutron absorber rods, used to manage reactor outlet temperature during normal operations and shutdown scenarios.

The following subsystem components are included within the RCSS systems:

- Control rod drive mechanism (CRDM)
- Neutron absorber rod (NAR)
- Rod guide tubes and bellows connected to graphite sleeves of the GCA
- CRDM motor housing and rotor housing
- Rod position and indication system (RPI)
- Control rod drive controller (CRDC) and power supply

The RCSS system is designed to provide excess negative reactivity to keep the reactor sub-critical during accident conditions with the most reactive rod from RCS and RSS each stuck in full out position. The industry standard codes and standards used to construct the RCSS system are provided in PSAR tables 6.4.10-1 and 6.4.10-2 and the process flow chart for RCSS is provided in PSAR figure 6.4.10-5.

System specific regulatory requirements applicable to the RCSS are as follows:

- 10 CFR 50.34(a)(5),
- 10 CFR 50.34(a)(8), and
- 10 CFR 50.35(a)

Additional guidance applicable for the evaluation of the RCSS is as follows:

- Interim Staff Guidance (ISG) DANU-ISG-2023-01 “Material Compatibility for non-Light Water Reactors,” (ML22203A175).
- Regulatory Guide 1.87, Revision 3, “Acceptability of ASME Code Section III, Division 5, 'High Temperature Reactors,'” (ML25176A084).
- Regulatory Guide 1.29, Revision 6, “Seismic Design Classification for Nuclear Power Plants,” (ML21155A003).

In addition to PDCs 1 through 6, the following system specific PDCs are applicable to the RCSS:

- PDC 10 - Reactor Design
- PDC-CDC 13 - Instrumentation and Control
- PDC 14 - Reactor Helium Pressure Boundary
- PDC-CDC 15 - Reactor Helium Pressure Boundary Design
- PDC-CDC 16 – Functional Containment Design
- PDC RFDC 26 and PDC-CDC 26 – Reactivity Control Systems
- PDC 28 – Reactivity Limits
- PDC-CDC 30 - Integrity of Reactor Helium Pressure Boundary
- PDC 31 - Fracture Prevention of Reactor Helium Pressure Boundary

6.4.10.1 Technical Evaluation

Safety Significance

The RCSS along with the associated subsystems is classified as safety related (SR). PSAR table 5.4-6 presents a list of LBEs that rely on RCSS for mitigation or prevention by inserting NARs into the core to add negative reactivity. While the inherent reactivity feedback in the system is sufficient to bring the reactor subcritical in the initial transient during an LBE, the rods perform a SR function of maintaining long-term subcriticality. If the RPS detects that the rods do not insert, it initiates a trip of the helium circulators to prevent the system from cooling down and returning to high power. Together these systems control heat generation during LBEs.

The RCSS system is also relied upon to demonstrate adequate DID to control reactivity and maintain HPB integrity during AOOs, DBEs, and BDBEs, as listed in PSAR table 5.5-1 and table 5.5-17, respectively. The DID function to control reactivity is provided by the same rods as the SR function but it is for rapid insertion during the initial transient as DID to the inherent reactivity feedback. The components within the system performing the NSRST HPB integrity function are the CRDM motor housing, rotor housing, and control rod bellows.

The RSFs required to be fulfilled by the RCSS during specific LBEs and the NSRST PSFs supported by the RCSS to demonstrate adequate DID are presented in PSAR section 6.4.10. The SR and NSRST portions of the RCSS are identified in PSAR section 6.4.10 with the overall system being categorized as SR.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR tables 6.4.10-1 and 6.4.10-2, which identify the design conditions, materials, and applicable codes and standards for the SR and NSRST portions of the RCSS, respectively. While the base materials for the SR components of the RCSS are identified, the weld materials are listed as to be determined (TBD) at the PSAR stage. The PSAR states that the governing safety function for the design and construction of these SSCs is RSF 1.1.2. The PSAR further indicates that RSF 1.1.2 for these components does not include requirements for demonstrating structural sufficiency since loss of structural integrity results in fail-safe configuration through NAR insertions by gravity. The PSAR further indicates that the NAR tubes, neutron absorber rod connecting chains, sprocket, and chain will be designed to

code philosophy consistent with the 2023 version of ASME BPVC Code, Section III, Division 5 to meet the rules and limits for load-controlled and deformation-controlled failures. The PSAR indicated that any deviations from ASME BPVC Code, Section III, Division 5 code philosophy will be justified as part of the OL application.

The NRC staff finds that construction code and materials for the SR components being listed as designed to code philosophy consistent with the 2023 version of ASME BPVC Code, Section III, Division 5 to be acceptable for the PSAR stage because the structural evaluation approach used by LME for the RCSS is consistent with the assessment methodology described in PSAR section 3.8.3 and is included in the material qualification plan. PSAR section 3.8.2 describes the construction code assessment process for considering material selection, geometric characteristics, and design and service conditions for SR SSCs. The acceptability of the structural evaluation of the metallic components is discussed in section 3.8.3 of this SE.

The NRC staff finds the level of detail provided to be appropriate for the CP stage given the preliminary nature of the design. The NRC staff will review the construction code and selection of the weld materials for SR components for the final RCSS design at the OL application stage .

For the NSRST components, the base, weld, and fastener materials are listed where applicable. PSAR table 6.4.10-2 identifies the code of construction for NSRST components of the RCSS. The HPB portion of the RCSS SSCs is constructed in accordance with the 2023 version of ASME BPVC Code, Section III, Division 5, Subsection HBA, as endorsed in RG 1.87, which was identified as a preliminary special treatment. The PSAR states that the RCSS stator housing cooling tubes are constructed to ASME B31.1, which adequately addresses all failure modes with appropriate margins. In addition, no augmentations to the baseline non-nuclear codes and standards were identified as necessary for the RCSS stator housing cooling tubes at this stage of design. The NRC staff finds the lack of augmentations to ASME B31.1 for the construction of the RCSS stator housing acceptable because the operating environment does not challenge the material of construction. The NRC staff finds the level of detail provided for codes and standards to be appropriate for the CP stage. The NRC staff will review the detailed design and construction specification of the NSRST components of the RCSS at the OL application stage.

PSAR section 6.4.10 states that at this stage of the design no supplementation to code requirements or planned testing for environmental compatibility assessment have been identified for the RCSS SSCs. The NRC staff will review the need for supplementation to code requirements, additional testing to address failure modes and modifications to the SSC characteristics, including any associated uncertainties to determine their impact on construction code and design requirements for the final RCSS design at the OL application stage..

The PSAR identifies a low temperature NSRST dissimilar weld between the CRDM motor housing and control rod nozzle in the RPV system as being constructed in accordance with ASME BPVC Code, Section III, Division 5, Subsection HBA. The construction code and environmental compatibility assessments of the weld are included in the materials qualification plan, which is assessed in section 3.8 of this SE.

The full scope of the LMGS QAPD, described in PSAR section 8.1, is applied to the SR RCSS SSCs. This program is designed to meet 10 CFR Part 50, Appendix B, and is appropriate for meeting the quality assurance requirements under PDC 1 for the SR aspects of the SR RCSS SSCs. The NSRST RCSS SSCs are designed, fabricated, constructed, and tested in accordance with the quality assurance program described in Part III of the LMGS QAPD and in PSAR section 8.1. Additional special treatments related to quality assurance will be specified at the OL application stage. Application of Part III of the LMGS QAPD at the CP stage is sufficient

as a preliminary special treatment to meet the quality assurance requirements under PDC 1 for the NSRST portions of the RCSS. This finding is based on the NRC staff's evaluation of the LMGS QAPD in the QAPD TR SE and in section 8.1 of this SE.

Based on its evaluation, the NRC staff finds that the PSAR provides adequate information on the codes and standards for the design, fabrication, construction and testing of the preliminary SR and NSRTS RCSS components to meet PDC 1.

Design Basis for Protection against Natural Phenomena *(PDC 2)*

The SR components of RCSS are physically located within the SR reactor support structures (RXST) and NSRST shield structure (SST) structures. Specifically, the SST provides protection from applicable natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions. The adequacy of the RXST and SST to protect the SR SSCs is evaluated in section 6.4.1 of this SE.

The NSRST RCSS components are located inside of the SST and will be designed to withstand the effects of natural phenomena consistent with the DBHL requirements identified in PSAR section 6.1.3 for NSRST components. The SR and NSRST RCSS components are designed to withstand seismic events and achieve performance levels corresponding to their designated seismic category as shown in PSAR table 3.7-2. As described in PSAR section 3.7, SR RCSS SSCs are classified as seismic category I, whereas NSRST RCCS SSCs are classified as seismic category IIA. The NRC staff finds this categorization consistent with the safety-significant functions of the RCSS because the SR and NSRST RCSS components are classified to their appropriate seismic category. The evaluation of seismic classification methodology and associated design requirements applied is documented in section 3.7.3 of this SE. The PSAR states that hazard protection strategies, processes, and methods which are part of design considerations will incorporate more detailed deterministic and probabilistic hazard assessments at the OL application stage. Based on its review, as discussed above, the NRC staff finds that the preliminary RCSS design features and the associated design development approach described in the PSAR are consistent with PDC 2. The NRC staff will review the detailed deterministic and probabilistic hazard assessments at the OL application stage to determine whether the final RCSS design meets the requirements of PDC 2

Fire Protection (PDC 3)

The NARs of the RCSS consist of pressed boron carbide absorber, which is a ceramic material with high thermal stability and is inherently non-combustible as well as non-fire sustaining. All other RCSS materials are designed and constructed of metallic materials, which are high-temperature resistant and are inherently non-combustible as well as non-fire sustaining. The energized stators of the RCSS drive motors are located outside the HPB and the power supplies are designed and constructed using non-combustible and fire-resistant materials to the extent practical. Further, the location of the RCSS on top of the RPV and in the GCA results in low combustible loading and limited fire hazard potential due to administrative controls associated with transient combustibles. Based on this information, the staff finds there is reasonable assurance that the design, location, and configuration of the RCSS minimizes the probability and effects of fires and explosions.

Further, the PSAR describes the fire protection program (FPP) and associated systems, which are designed to meet the requirements of 10 CFR 50.48(a)(1) and PDC 3. The fire protection system architecture is described in PSAR section 8.6, and PSAR section 1.1.4.3.9 describes key provisions for protecting safety significant SSCs from fire hazard.

Based on the preliminary design features of the RCSS and programmatic commitments described in PSAR sections 1.1.4.3.9 and 8.6, the staff finds that the fire protection measures for safety-significant SSCs of the RCSS meet the requirement of PDC 3 to minimize the probability and effects of fires and explosions.

Environmental and dynamic effects design bases (PDC 4)

The NRC staff reviewed PSAR section 6.4.10 on the compliance of the RCSS preliminary design with the requirements in PDC 4 to accommodate the effects of and to be compatible with the environmental conditions and found that these requirements are satisfied by performing the environmental compatibility assessment and developing the material qualification plan to mitigate the applicable material degradation modes identified from the material compatibility assessment, as described in PSAR section 3.8. The methodology described in PSAR section 6.1.1 is applied for the evaluation of dynamic effects from internal and external hazards. The safety-related RCSS material qualification plan is identified as part of the R&D program. The acceptability of the material qualification plan, of which the R&D program is a part, is evaluated in section 3.8 of this SE.

For the requirements on protection against dynamic effects, PSAR section 6.4.10 states that as the design matures, the sources of dynamic hazards to the RCSS will be assessed systematically and for dynamic hazards determined to be credible, analyses will be performed to demonstrate adequate RCSS protection. Design measures to mitigate credible dynamic effects include a detailed compatibility assessment for material wear, swelling of the B_4C as well as loading considerations in material qualification for NAR from internal gas release of the B_4C during service lifetime. The PSAR further identifies the fuel handling system (FHS), cross vessel (CV) and reactor pressure vessel (RPV) as potential sources of dynamic hazards to the RCSS. LME states that the credibility assessment, final design approach, supporting analyses, and demonstration that dynamic effects have been appropriately addressed will be documented in the FSAR to be submitted with a future OL application. The NRC staff finds LME's approach to ensure adequate protection of the RCCS from dynamic effect acceptable because it provides a structured, commitment-backed process to evaluate dynamic hazards as the design matures.

Based on the above information, the NRC staff finds that the preliminary design of the RCSS meets the requirements of PDC 4. Staff will evaluate the materials qualification program as well as the final design at the OL application stage.

Sharing of structures, systems, and components (PDC 5)

The RCSS components are not shared between the reactor modules. Each reactor module has its own dedicated RCSS. However, the PSAR indicates that the RCSS components share a foundation system with multiple safety-significant SSCs. The adherence to PDC 5 within the context of a shared foundation system is described in PSAR section 3.7.6.3 and the acceptability of such sharing is documented in section 3.7 and section 6.4.1 of this SE. Based on its review, the NRC staff concludes that there is sufficient detail in the PSAR regarding the preliminary RCSS design to demonstrate that safety-significant SSCs will not be shared among nuclear power modules and that the sharing of the common foundation will not significantly impair the ability of the RCSS to perform their safety functions, commensurate with the safety-significance of the functions to be performed in accordance with PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The functional testing of the RCSS will ensure the operability and performance of its safety-significant components under conditions as close as practical to those expected during LBEs. PSAR section 6.4.10 states that the RCSS meets the testing portion of PDC 6 through pre-service and in-service requirements and that the functional testing to ensure operability and performance of the active portions of SR RCSS will be performed in accordance with the maintenance program as described in PSAR section 8.3. Planned testing includes tests for rod insertion and movements. The evaluation of the maintenance program for functional testing is described in section 8.3 of this SE.

The PSAR states that periodic inspections of NSRST RCSS SSCs will be conducted under the RAM program, as described in section 8.10 of the PSAR. The RCSS design provides access for implementing remote pre-service and in-service requirements. The potential MANDE requirements include SSC health monitoring, water chemistry control for stator cooling, and helium side chemistry monitoring. Because the design provides access and proposed monitoring addresses credible degradation mechanisms, the staff finds the preliminary monitoring measures provide reasonable assurance that the functional capability of the RCSS will be maintained. The PSAR notes that the operability of the RCSS as a whole, including associated systems, for AOOs, DBEs, and DBAs is performed as part of pre-operational testing and startup testing described in PSAR sections 12.1.1.2 and 12.1.1.3, respectively. The evaluation of the pre-operational testing and startup testing are described in section 12.1 of this SE.

The NRC staff will evaluate the RAM and maintenance programs as part of the OL application review, including the basis and acceptability of the departure from the scope of ASME Section XI-2 as endorsed in RG 1.246 (SE section 8.2) which includes SSCs that perform functions requiring special treatment for defense in depth adequacy Based on the discussion above, the NRC finds that the preliminary design of the RCSS meets the requirements of PDC 6..

Reactor Design (PDC 10)

The NRC staff reviewed PSAR section 6.4.10, which states that PDC 10 is a higher-level requirement that is met for the RCSS via application of PDC-CDC 26 which is aligned with performance of NSRST PSF 1.1.3 to ensure that SARRDLs and HPB design limits are not exceeded and safe shutdown is achieved and maintained during AOOs. The PSAR further indicates that appropriate design margins that preclude exceeding SARRDLs are inherent in the appropriately designed safety, quality, and seismic classification of the RCSS, the application of appropriate codes and standards commensurate with the RSFs and PSFs for the RCSS, and the overall selection of materials with consideration to operating temperatures and pressures that the RCSS components would experience during any normal operating conditions or AOOs. The NRC staff notes that SARRDLs were not included in the CP application, but the NRC staff reviewed the PSAR dose consequence results for AOOs for a preliminary assessment, considering margin to the F-C target.

The NRC staff finds the discussion acceptable because it is consistent with the discussion in the NRC-approved Xe-100 LTR PDC, Rev. 3, and the updated basis for the PDC 10 discussed in PSAR section 5.3. As discussed in the basis for PDC 10 in the LTR, PDC10 is a design limit that supports acceptance criteria in PDC-CDC 26. PDC-CDC 26 is aligned with performance of NSRST PSF 1.1.3 to ensure that the SARRDLs are not exceeded and that safe shutdown is achieved and maintained during the AOOs. Therefore, PDC 10 requirements are satisfied by

meeting the requirements in PDC-CDC 26. The evaluation for compliance of the RCSS design to PDC-CDC 26 is presented later in this section.

Reactivity Control Systems (PDC RFDC 26)

The RCSS design includes two mechanisms to initiate a reactor trip of the RSS and RCS rods; RPS trip and manual trip. These trips remove electric power from the CRDM motor and both RCS and RSS banks insert under the force of gravity to achieve and maintain shutdown. This reactor trip capability ensures performance of RSF 1.1.2, "Maintain long-term subcriticality," with no reliance on operator action or electrical power. PSAR table 3.1.6-2 presents reactivity worths for the RCS and RSS rod banks and shows that the rods have enough reactivity worth to achieve and maintain safe shutdown with one rod assumed out for each rod bank. The details of the RPS control functions to support PDC-RFDC 26 are provided in PSAR section 6.4.13.

Based on the description of the RCS and RSS rod banks and their capability to perform the required RSF 1.1.2, as discussed above, the NRC staff finds that the preliminary design of the RCSS meets the requirements of PDC-RFDC 26.

Reactivity Control Systems (PDC CDC 26)

PDC-CDC 26 is aligned with the performance of NSRST PSF 1.1.3, "Control reactivity with moveable poisons." The RCSS is designed to insert neutron absorbing rods (NAR) in stepwise, controlled movements into the core at a rate sufficient to ensure safe shutdown is achieved and maintained during the AOOs without exceeding the SARRDLs and HPB design limits. PSAR table 3.1.6-2 shows reactivity worths for the RCS and RSS rod banks. The average rod speeds, distance covered per step, and other pertinent design information related to performing the PSF 1.1.3 are provided in PSAR section 6.4.10.2. The details of the IPS and DCS control functions to support PDC-CDC 26 are provided in PSAR section 7.3.11 and 7.3.12, respectively.

Based on the description of the RCS and RSS rod banks and their capability to perform the required PSF 1.1.3 and meet the PSAR dose consequence results for AOOs, as discussed above, the NRC staff finds that the preliminary design of the RCSS meets the requirements of PDC-CDC 26.

Instrumentation and Control (PDC CDC 13)

The instrumentation associated with RCSS is designed to monitor the position of control rods and other variables related to reactivity control within their anticipated ranges during normal operations and AOOs. The RCSS subsystems supporting this function are the CRDM, rod position indication, and CRDC. The instrumentation and controls associated with the RCSS further ensure the ability to achieve and maintain safe shutdown during AOOs through interface with plant protection and control systems. The details of the IPS control actions to support the PDC CDC 13 are provided in PSAR section 7.3.11 and are evaluated in section 7.3.11 of this SE.

Based on the description of the instrumentation system provided for the RCSS and its capability to provide monitoring for variables and controls associated with the performance of the system, the NRC staff finds that the preliminary design of the RCSS meets the requirements of PDC-CDC 13.

Reactor Helium Pressure Boundary (PDC 14), Reactor Helium Pressure Boundary Design (PDC-CDC 15) & Functional Containment Design (PDC-CDC 16)

The NRC staff reviewed PSAR section 6.4.10, which states that the pressure retaining part of the RCSS, such as control rod nozzles, CRDM housings, rotor housings, and their sealed connections support PDC 14, PDC-CDC 15, and PDC-CDC 16 through use of the appropriate ASME codes and standards as the primary means of assuring HPB integrity and support performance of NSRST PSF 2.7, "Maintain the HPB pressure integrity," and the associated sub-PSFs. The use of industry codes and standards in the selection of HPB material for the RCSS considers operating design conditions, as shown in PSAR tables 6.4.10-1 and 6.4.10-2. Further, features to minimize leakage are used where appropriate and include minimizing penetrations and connections between HPB components and appropriate design considerations for seals, welds, and flange connections. The PSAR states that design of connections and other leakage minimization features will be finalized as appropriate and submitted as part of a future OL application. The NRC staff finds this acceptable because the preliminary design details to assure HPB integrity and support performance of NSRST PSF 2.7 are provided in the PSAR. The NRC staff will review the final design of the connections and other leakage minimization features at the OL application stage.

Based on the description of the industry standard codes and standards used in the RCSS design and description of the design philosophy applied to minimize leakage within the pressure retaining part of the RCSS, the NRC staff finds that the preliminary design of the RCSS meets the requirements of PDC 14, PDC-CDC 15, and PDC-CDC 16.

Reactivity Limits (PDC 28)

The operation and performance of the RCSS ensure compliance with PDC 28. Specifically, the key design features of the RCSS, discussed below, ensure performance of RSF 1.1.2, "Maintain long-term subcriticality," RSF 1.1.2.1, "Maintain long-term subcriticality during safe shutdown conditions," and NSRST PSF 1.1.3, "Control reactivity with moveable poisons." Limitations on the amount and rate of reactivity increases are implemented through the design of the rods and the CRDC system along with its interfaces with other plant control systems. The CRDC system provides rod position feedback to the DCS. The average speeds for rod movements along with the rest period for the NAR are described in PSAR section 6.4.10.1. The rest period allows the control system to confirm successful completion of the desired reactivity change. During RPS reactor trips, electrical power is removed from CRDC motors, and the NARs are fully inserted to ensure safe shutdown without creating any additional damage to the HPB or impairing the core cooling capability.

Based on the description of the systems and interfaces of RCSS and the discussion on their capability to perform the associated RSFs and NSRST PSF without any additional damage to the HPB or impairing the core cooling capability, the NRC staff finds that the preliminary design of the RCSS meets the requirements of PDC 28.

Integrity of Reactor Helium Pressure Boundary (PDC-CDC 30)

The PSAR states that the portions of the RCSS that are part of the HPB are designed to prevent helium leakage beyond normal makeup capability. The intent of PDC-CDC 30 is met with appropriate codes and standards, as discussed in PDC 1, and design features discussed to meet PDC 14, PDC-CDC 15, and PDC-CDC 16. The detection and isolation of helium leakage is met through application of PDC-CDC 13 as a design requirement applied to the interfacing systems such as NIS and IPS. NIS is a SR system described in PSAR section 6.4.14 and IPS is a NSRST system described in PSAR section 7.3.11. Consistent with PDC-CDC 30, the NIS is designed to sense abnormal helium egress from the HPB by detecting pressure changes and

supplying signals to the IPS. The IPS system is designed to automatically isolate sources of helium leakage when conditions requiring the isolation are detected by the NIS.

Based on the description provided in the PSAR regarding the codes and standards applied to the HPB components of the RCSS, design features discussed to prevent helium leakage, as well as the design requirements on the interfacing systems (such as NIS and IPS) to detect and isolate any HPB leakage, the NRC staff finds that the preliminary design of the RCSS meets the requirements of PDC-CDC 30.

Fracture prevention of reactor helium pressure boundary (PDC 31)

The design of the HPB portions of the RCSS is performed with considerations of service temperatures, degradation of material properties, creep, stress rupture, fatigue and other conditions for all operating modes. The uncertainties associated with the material properties as well as the effects of irradiation and other stressors are considered in the design. The stresses experienced by the pressure retaining parts of the RCSS are determined for the range of credible transients and applied through the appropriate use of codes and standards and other design constraints to meet PDC 14, PDC-CDC 15, and PDC-CDC 16. The design conditions, materials, and codes of construction for the RCSS components are provided in PSAR tables 6.4.10-1 and 6.4.10-2. The PSAR provides description of seismic category applied to SR and NSRST RCSS components. The acceptability of seismic classification is discussed in the evaluation of PDC 2 for the RCSS. The uncertainties in material properties and the environmental compatibility assessment are addressed through the material qualification plan. The NRC staff finds this to be acceptable since the structural evaluation approach for the RCSS SSCs follows the assessment methodology described in PSAR section 3.8.3 and is included in the material qualification plan.

Based on the description provided in the PSAR regarding the codes and standards, design features, and the assessment methodology for the RCSS, the NRC staff finds that the preliminary design of the RCSS meets the requirements of PDC-CDC 31.

Safety Functions

PSAR section 6.4.10.1 states that the SR portion of the RCSS is designed to meet the following RSFs:

RSF 1.1.2, “Maintain long-term subcriticality”
RSF 1.1.2.1, “Maintain long-term subcriticality during safe shutdown conditions”

The RCCS applies the plant-level PDC-RFDC 26, ‘Reactivity control systems,’ to support performance of RSFs 1.1.2 and 1.1.2.1.

The following system-level SRDCs are applied to the RCSS to ensure successful performance of RSF 1.1.2.1:

- The RCSS neutron absorbers are designed with a sufficient reactivity worth to ensure safe shutdown in DBEs and DBAs.
- The RCSS is designed to insert negative reactivity within a sufficient period to ensure safe shutdown during a DBE or DBA.

The PSAR states that the RCSS is also designed to meet the following PSFs, which are required to be performed during specific LBEs:

PSF 1.1.3, "Control reactivity with moveable poisons"
PSF 2.7, "Maintain HPB pressure integrity"

The following system-level CDCs are applied to the RCSS to ensure successful performance of PSF 1.1.3:

- CDC for PSF 1.1.3.1 – The RCSS provides sufficient reactivity worth capability to support preventing occurrence of reactor trips during AOOs.

The following system-level CDCs are applied to the RCSS to ensure successful performance of PSF 2.7:

- CDC for PSF 2.7.1 – The RCSS, as part of the reactor system, is designed to minimize the likelihood of small depressurization AOOs.
- CDC for PSF 2.7.2 – The RCSS, as part of the reactor system, is designed to minimize the likelihood of medium depressurization AOOs.
- CDC for PSF 2.7.3 – The RCSS, as part of the reactor system, is designed to minimize the likelihood of large depressurization AOOs.

The NRC staff finds that the preliminary RCSS design supports PSF 1.1.3 through the application of PDC-CDC 26. The performance of PSF 2.7 is supported through application of PDC 1, PDC 14, PDC-CDC 15, and PDC-CDC 16 by use of the appropriate codes and standards, commensurate with their safety functions.

As part of the regulatory audit (MLXXXXX), the NRC staff inquired about RSF 1.4.1, "Maintain core geometry," not being listed in the PSAR since the description of the control rod guide tubes and bellows provided states that these guide tubes are each connected to a bellows expansion joint below the RPV head and transition into carbon graphite guide sleeves of the radial reflector. Based on the information presented in the PSAR, and verified in audit discussions with LME, the NRC staff found that the control rod guide tubes do not support core system geometry and are not relied upon to provide path for rod insertion or to prevent any damage to core structures such that rod insertion can be achieved. Therefore, the NRC staff concluded, based on the description provided in the PSAR section 6.4.6, that RSF 1.4.1 is not applicable to the preliminary RCSS design. The NRC staff will further confirm the applicability of RSF 1.4.1 to the RCSS at FSAR stage as detailed design information is available.

The NRC staff finds the description of the safety functions to be consistent with the guidance for the content of CP applications found in section C.6.4, "Descriptions for SR SSCs," of NEI 21-07 as endorsed by RG 1.253, since the features to demonstrate system capability and reliability for prevention and mitigation are reflected in the description of RCSS components provided in PSAR section 6.4.10.1 and the system performance and operation discussion provided in PSAR section 6.4.10.2.

First-of-a-Kind (FOAK) Testing

Criterion XI, "Test Control," of Appendix B to 10 CFR Part 50 states, in part, that, "[a] test program shall be established to assure that all testing required to demonstrate that SSCs will

perform satisfactorily in service is identified and performed in accordance with written test procedures, which incorporate the requirements and acceptance limits contained in applicable design documents."

The PSAR did not identify any RCSS component as incorporating first-of-a-kind (FOAK) technology, design or configuration and thus did not identify any FOAK special treatment tests to demonstrate capability of RCSS in performing necessary RSF and NSRST PSF or meeting regulatory requirements. As part of the regulatory audit, the NRC staff requested clarification on why certain aspects of the RCSS design which appear to incorporate novel features were not identified as FOAK. These aspects included CRDM design with chain type operation, segmented NARs with mechanical connection, stacked boron carbide ring geometry inside cladding tubes, and the dual bank RCSS design.

LME, through the audit, responded that:

- The segmented NAR is not unique to the Xe-100 design. Segmented rod designs have been used in other high temperature gas reactors (HGTRs) including the Japanese high temperature test reactor (HTTR) and the Chinese 10-megawatt high temperature gas-cooled test module (HTR-10). The clearances between rod outer diameter and graphite core sleeve diameter allow for any rod section bowing.
- The CRDM design with chain type operation was previously used in Fort St. Vrain. From an operating perspective both Xe-100 and Fort St. Vrain CRDMs incorporate fully energized motors and brake assemblies to maintain rod position with minor design differences between the two designs with chain sprocket.
- The stacked boron carbide ring has been used in other HGTRs including the HTTR and HTR-10, with both designs reporting five compacts being used inside each segment. The design of Xe-100 NARs also ensures clearance between stacked boron carbide and the cladding to allow for thermal expansion and material form degradation over time.
- The dual bank rod architecture has been used in the advanced gas reactors (AGRs) in the UK which utilize "bulk rods" for long term reactor shutdown while "regulating" rods are driven automatically to balance power. The design of RCSS dual architecture is similar in purpose to the AGRs.
- The materials used in the NARs metallics and the connecting chain are limited to 800H and Alloy 625, which have been used previously in other HGTRs, and the helium coolant with graphite moderated environment is well established in HGTR designs.

The NRC staff reviewed the information provided in the audit and confirmed its applicability to the Xe-100 design based on publicly available information. Based on the comparison of the design information in the PSAR and the publicly available information, the NRC staff concluded that LME's determination that the components and design features in question do not warrant FOAK special treatment is reasonable.

The NRC staff will evaluate the continued validity of the FOAK treatment at the OL application stage as the RCSS design matures and detailed design information is available.

Technical Specifications

The RCSS has been identified to have technical specifications based on the preliminary design and the associated preliminary safety analysis, with additional details provided in Part IV of the CP application, "Technical Specifications." The proposed list of preliminary LCOs identified in

the proposed technical specifications includes a minimum number of rods per bank to support RSF 1.1.2, "Maintain long-term subcriticality," to be fulfilled during specific LBEs. At the CP stage, the PSAR identified no safety limits (SLs) or surveillance requirements for the RCSS.

Based on the review of the information provided in the CP application, as discussed above, the NRC staff finds the information provided meets 10 CFR 50.34(a)(5). The NRC staff finding is based on variables, conditions, or items that are likely to become subjects of technical specifications for the RCSS being identified. Consistent with 10 CFR 50.34(b)(6)(vi), the NRC staff will evaluate the proposed technical specifications -when detailed design information for the RCSS is available at the OL application stage.

Research and Development

10 CFR 50.34(a)(8) requires identification of any facility structures, systems, or components that necessitate research and development to confirm the adequacy of their design.

Table 13.2-1 of PSAR Chapter 13 identifies RCSS as scoped within the proposed LME R&D program to resolve the safety question associated with the performance of its metallic materials in the Xe-100 environmental operating conditions. The evaluation of the SR metallics of the RCSS in the R&D program is discussed in Chapter 13 of this SE.

Additional Testing

During the PSAR review, the NRC staff identified potential concerns related to the NAR hanging up within the channel based on the block type nature of the construction and impacts of manufacturing tolerances and shifting that may occur in the GCA.

LME, in the regulatory audit, provided measurements of outer dimensions of the NARs and the inner dimension of the sleeves to show sufficient space to not bind during thermal swelling or other distortions. LME pointed to data from UK AGR, Japanese HTTR, and Chinese HTR-10 reactors, which employ segmented rod design in similar configuration. With regards to the shifting of the Graphite, LME stated in the audit that it will be considered as part of the R&D program for the GCA to ensure RSF 1.4.1, "Maintaining Core Geometry," is met. The audit response further pointed to development of rod verification methodology and physical testing to ensure rod insertion for investment protection purposes. This rod verification methodology to evaluate the performance of rod insertion into the graphite channels and ensure sufficient margin to failure is maintained throughout GCA operational life is described in PSAR section 6.4.4. PSAR chapter 12 discusses the pre-operational and start-up test programs, which include RCSS testing.

Based on the review of the RCSS design presented in Section 6.4.10 of the PSAR and confirmatory discussions from regulatory audit, the NRC staff finds that the plan to address the rod insertion concerns through planned testing and verification activities is acceptable because it provides NRC staff with reasonable assurance that technical and design information related to NAR movement and insertion can be left for later consideration at the OL stage, and that the planned testing and verification activities are reasonably designed to resolve such potential concerns, in accordance with 10 CFR 50.35(a).

6.4.10.2 Conclusion

Based on the review described above, the NRC staff concludes that the information in PSAR section 6.4.10 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP. Specifically, the information provided:

- meets requirements of 10 CFR 50.34(a)(3)(i) for the RCSS by describing the preliminary design identifying the applicable PDCs.
- meets the requirements of 10 CFR 50.34(a)(3)(ii) by describing how the RCSS design satisfies the applicable PDCs.
- meets 10 CFR 50.34(a)(3)(iii) by providing information relative to materials of construction, general arrangement, and approximate dimensions in the RCSS, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.
- meets 10 CFR 50.34(a)(4) by describing preliminary and planned analyses and the safety margin for safety functions provided by the RCSS design.meets 10 CFR 50.34(a)(5) by identifying key RCSS variables and conditions as probable subjects of technical specifications.
- meets 10 CFR 50.34(a)(8) by identifying the research and development needs for RCSS to confirm the performance of their metallic materials in the Xe-100 environmental operating conditions.
- meets 10 CFR 50.35(a) with respect to NAR movement and insertion by providing reasonable assurance that technical and design information can be left for later consideration at the OL stage, that the planned testing and verification activities are reasonably designed to resolve any safety questions, and that such safety questions will be satisfactorily resolved at or before the latest date for completion of construction.

6.4.10.3 References

- Interim Staff Guidance (ISG) DANU-ISG-2023-01 “Material Compatibility for non-Light Water Reactors,” Accession Number: ML22203A175

- Regulatory Guide 1.87, Revision 3, “Acceptability of ASME Code Section III, Division 5, 'High Temperature Reactors'” Accession Number: ML25176A084

- Regulatory Guide 1.29, Revision 6, “Seismic Design Classification for Nuclear Power Plants.” Accession Number: ML21155A003

6.4.11 Reactivity Cavity Cooling System

PSAR section 6.4.11 describes the reactor cavity cooling system (RCCS). The RCCS provides passive and active heat transfer from the RPV to the atmosphere. The RCCS comprises the standpipe functional area (SFA) and the process functional area (PFA).

The SFA is the passive SR core and reactor cavity heat removal subsystem utilized in DBAs and DBEs. The SFA consists of gooseneck vents to atmosphere for vapor egress and emergency filling lines, boil off supply tanks, risers, downcomers, riser headers, and downcomer headers. PSAR figure 6.4.11-3 shows the components of the SFA. The PFA is the active NSRST subsystem that operates with the SFA to provide active reactor cavity heat removal for a range of LBEs. The PFA contains pumps, air cooled heat exchanger banks, instrumentation, and supply and return piping. PSAR table 6.4.11-3 lists the industry consensus codes and standards used to construct the RCCS SFA and PFA.

PSAR section 6.4.11.1.1 provides a detailed overview of the SFA and the PFA portions of the RCCS. PSAR figure 6.4.11-2 provides the process flow diagram for the RCCS SFA and PFA subsystems. The key RCCS design features are summarized below.

1. Each reactor module incorporates two independent SR SFA trains. Every train includes 25 standpipes, each consisting of downcomer and riser sections (14-inch OD) connecting the toroidal downcomer header to the riser header. Two boil-off tanks are connected above each downcomer header. Both the boiloff tanks and the riser headers vent to atmosphere outside the SST through the gooseneck vents.
2. All SR SFA components are located within the NSRST SST, except the gooseneck vents, which penetrate the SST roof and are external to the SST.
3. SR SFA subsystems are hydraulically connected to a single shared NSRST PFA subsystem via supply and return piping. Piping segments within the SST (connecting boil-off tanks and riser headers) are classified as SR, while piping located outside the SST and interfacing with the PFA system is classified as NSRST.
4. The total water inventory, comprising four boil-off tanks, two riser headers, and standpipe sections above the RPV top head, is sized to support 72 hours of passive cooling. Defense-in-depth makeup options (NIFP, NIPW, CDCN, offsite water, and direct hose supply through boil-off vents) allow passive cooling beyond 72 hours.
5. The highest elevation of the supply and return piping is located above the boil-off tank water level. This arrangement prevents loss of 72-hour passive inventory following leak or break in PFA portion of the RCCS by enabling isolation of the SFA once water level drops below this highest piping elevation.
6. All NSRST PFA components, including three cooling water pumps, six air-cooled heat exchangers (with total 150% capacity), supply and return piping, and isolation valves are located outside the SST and are shared across all four modules.
7. The NSRST isolation valves are motor-operated with manual override capability. Their primary purpose is to isolate an affected SFA train in the event of a detected leak or break. Valve closure prevents inventory loss from unaffected modules by isolating the compromised SFA while retaining PFA functionality for the remaining modules.
8. In active mode, PFA pumps circulate water through both SFA and PFA. Cold water is provided to boil-off tanks and downcomer headers; heated water returning from riser

headers is cooled by the PFA air-cooled heat exchangers. Active cooling maintains reactor cavity concrete temperatures below 65 °C during normal operation and AOOs. PSAR table 6.4.11-2 shows the process parameters for the RCCS in active mode.

9. Loss of active cooling causes transition to passive RCCS operation. Without PFA supply, water in the riser standpipes reaches near-boiling conditions, generating steam that rises to riser headers and vents through the gooseneck vents. Boil-off tanks continuously feed the downcomer piping via natural circulation. No operator action or active component operation is required in passive mode.

System specific regulatory requirements applicable to the RCCS are as follows:

- 10 CFR 50.34(a)(5), and
- 10 CFR 50.35(a)(2)

In addition to PDCs 1 through 6, the following system specific PDCs are applicable to the RCCS:

- PDC 10 - Reactor Design
- PDC-CDC 13 - Instrumentation and Control
- PDC-RFDC 34 and PDC-CDC 34 – Residual Heat Removal

System specific regulatory guidance applicable to the RCCS is as follows:

RG 1.203, “Transient and Accident Analysis Methods,” (ML053500170)

6.4.11.1 Technical Evaluation

Safety Significance

The SFA portion of the RCCS is classified as safety related (SR). The NRC staff reviewed PSAR table 5.4-8, which presents a list of LBEs that rely upon the SFA portion of the RCCS for mitigation or prevention by removing heat from the reactor core and reactor cavity using passive boil-off water from RCCS inventory. The SFA portion of the RCCS is designed to mitigate events such that:

- Consequences of DBEs are kept within the F-C target
- DBAs that are derived from the selected DBEs meet the dose limits specified in 10 CFR 50.34.
- BDBEs with consequences greater than the 10 CFR 50.34 dose limits do not become DBEs.

The PFA portion of the RCCS is relied upon to demonstrate adequate DID to actively control heat removal from the reactor cavity during AOOs, DBEs, and BDBEs, as listed in PSAR table 5.5-11.

The NRC staff finds the SFA portion of the RCCS is appropriately categorized as a SR system as it is relied upon to mitigate consequences of the DBEs and BDBEs by performing its RSFs as well is to mitigate consequences of the DBAs derived from the select DBEs. Similarly, the NRC staff finds that the PFA portion of the RCCS is appropriately categorized as a NSRST system as it is relied upon to mitigate consequences of the AOOs, DBEs, and BDBEs.

Compliance with Applicable PDCs

The RCSS is required to meet the following PDCs:

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR table 6.4.11-3, which identifies the applicable design codes and standards for the RCCS while PSAR tables 6.4.11-4 and 6.4.11-5 identify the design conditions, materials, and construction codes for the SFA and PFA components, respectively. SR components of the RCCS are designed and manufactured according to the 2023 version of ASME Code, Section III, Division 5, with type 304 stainless steel as base material. All welds of SR RCCS components will be compliant with ASME Code, Section III, Division 5, Subsection HC, Subpart A (Class B) and Subsection HF, Subpart A for the supports. PSAR table 6.4.11-5 states that the NSRST components of the RCCS will be made of 304 or 304L stainless steel. All welds of NSRST components will be compliant with either ASME VIII.1 or ASME B31.1. PSAR table 6.4.11-5 identifies the code of construction as either ASME VIII.1 or ASME B31.1 for each NSRST component of the RCCS.

PSAR section 6.4.11.1.2 further states that augmented weld integrity measures beyond baseline ASME VIII.1 and B31.1 are not needed as special treatments for NSRST RCCS because of the low pressure and temperature conditions, absence of energetic chemical reactions, and limited scope of the PFA's NSRST safety function. The PSAR further states that this approach will be confirmed as design matures and will be appropriately reflected in the FSAR.

The full scope of the LMGS QAPD, described in PSAR section 8.1, is applied to the SR RCCS SSCs. This program is designed to meet 10 CFR Part 50, Appendix B, and is appropriate for meeting the quality assurance requirements under PDC 1 for the SR aspects of the SR RCCS SSCs. The NSRST RCCS SSCs are designed, fabricated, constructed, and tested in accordance with the quality assurance program described in Part III of the LMGS QAPD and in PSAR section 8.1. Additional special treatments related to quality assurance will be specified at the OL application stage. Application of Part III of the LMGS QAPD at the CP stage is sufficient as a preliminary special treatment to meet the quality assurance requirements under PDC 1 for the NSRST portions of the RCCS. This finding is based on the NRC staff's evaluation of the LMGS QAPD in the QAPD TR SE and in section 8.1 of this SE

Based on the information provided in the PSAR, the NRC staff finds that the applicant's selection and use of the codes and standards for the design, fabrication, construction and testing of the SR and NSRTS RCCS components is in accordance with PDC 1. The NRC staff will confirm special treatments applied for the NSRST RCCS component welds at the OL application stage.

Design Basis for Protection against Natural Phenomena (PDC 2)

The NRC staff reviewed PSAR section 6.4.11.1.2, which states that all the SR components of the RCCS, except gooseneck vents, are physically located within the SR reactor support structures (RXST) and NSRST shield structure (SST). The adequacy of RXST and SST to protect the SR SSCs is evaluated in section 6.4.1 of this SE. The SR gooseneck vents are located on the top of boil-off tanks and riser headers and are directed to the atmosphere outside of the SST on the top of the RB. The PSAR states that these vents are designed to withstand the design basis hazard levels described PSAR section 6.1.1 and the loads identified in PSAR table 6.1-1. The PSAR confirms that the final design will include missile barriers or other protective features to protect the gooseneck vents from missiles generated by natural phenomena.

PSAR section 6.4.11.1.2 states that NSRST RCCS components are located outside of the SST and will be designed to withstand the effects of natural phenomena consistent with the DBHL requirements identified in PSAR section 6.1.3 for NSRST components. As described in PSAR section 3.7, SR RCCS SSCs are classified as seismic category I, whereas NSRST RCCS SSCs are classified as seismic category IIA. PSAR section 3.7.6.3.2 describes that the foundation design for both SR and NSRST foundations considers differential settlement loads in combination with other loads in accordance with the applicable load combinations. As described in the staff's evaluation in SE section 3.7.1, LME ensures adequate performance of the foundations and systems that may span across multiple foundations by meeting deflection limits in ASCE 7-16, ACI-318-14, and ACI 349-13 associated with serviceability requirements. Furthermore, SE section 3.7.1 notes that LME ensures adequate protection against differential settlement by conducting the foundation related work in conformance with LME's quality assurance program and verifying the capacity of individual piles (where part of the foundation system) by load tests. Based on the evaluation in SE section 3.7.1, the NRC staff concludes that LME's proposed approach provides reasonable assurance that the foundations will remain functional when subjected to differential settlement loads and associated load combinations without impacting performance of the supported SR and NSRST SSCs including the RCCS vents.

Based on its review, the NRC staff finds that the PSAR provides sufficient detail regarding the RCCS design features and the associated design development approach to demonstrate that both SR and NSRST RCCS components are protected against the effects of natural phenomena in accordance with PDC 2.

Fire Protection (PDC 3)

The PSAR describes a comprehensive FPP and associated systems designed to meet the requirements of 10 CFR 50.48(a)(1) and PDC 3. The FPP described in PSAR section 8.6 incorporates programmatic elements such as Fire Hazards Analysis (FHA), Safe Shutdown Analysis (SSA), administrative controls, personnel training, and maintenance programs. PSAR Section 8.6 states the operational FPP will conform to Regulatory Guide 1.189 and relevant provisions of NFPA 804, ensuring post-fire safe shutdown capability and mitigation of radiological release risks. Furthermore, PSAR section 1.1.4.3.9 describes key provisions for protecting safety significant SSCs that include:

- Passive fire barriers and compartmentalization to isolate fire zones and confine fire effects on redundant safety-significant SSCs (PSAR section 1.1.4.3.9);
- Active fire suppression systems, including automatic water spray/mist systems, clean agents, and manual standpipes for firefighting (PSAR section 1.1.4.3.9);

- Fire detection and alarm systems for early detection, monitoring, and alarm notification (PSAR section 1.1.4.3.9);
- Reliable fire water supply infrastructure, including storage tanks, pumps, and hydrants (PSAR section 1.1.4.3.9).

Based on the preliminary design features and programmatic commitments described in PSAR sections 1.1.4.3.9 and 8.6, the staff finds that the fire protection measures for SR SSCs are consistent with the intent of PDC 3 to minimize the probability and effects of fires and explosions. The staff's evaluation of the FPP is described in SE section 8.6.

Environmental and dynamic effects design bases (PDC 4)

The NRC staff reviewed PSAR section 6.4.11.1.2, which states that the structural evaluation methodology described in PSAR section 3.8 is applied to design, construction, material selection, and environmental compatibility assessment of the SR and NSRST RCCS metallic components. The methodology described in PSAR section 6.1.1 is applied for the evaluation of dynamic effects from the internal and external hazards. For the NSRST RCCS, the DBHL requirements in PSAR section 6.1.3 may be applied to prevent the adverse impact of NSRST RCCS on the SR RCCS performance. The PSAR states that as the design matures, the dynamic hazards from failure of co-located equipment such as FHS, CV, RPV, and RCSS will be systematically assessed.

PSAR section 6.4.11.1.1 describes that the SR boil-off tanks are secured and anchored to the steel framing of the RXSTs. The PSAR further states that the design of the support structure and steel framing that maintain the standpipe assembly in position and proper spacing are not complete. The standpipes themselves are directly affixed to the RXST steel framing. The RXST provides structural support for ASME Section III, Division 5 supply and return piping, RCCS standpipes, downcomer headers, riser headers, and tanks.

PSAR table 6.4.11-2 provides RCCS process parameters during active operation. PSAR tables 6.4.11-4 and 6.4.11-5 present the bounding design conditions for the RCCS SR and NSRST components during passive operation, as well as the selected construction materials and construction codes for the RCCS SSCs. As discussed in PDC 1, the selected construction materials and associated codes are appropriate for the RCCS design conditions.

The PSAR also addresses how the SR RCCS will account for the following design issues:

- Thermal fatigue caused by temperature fluctuations from potential two-phase flow and boiling instabilities
- Impact of fouling on RCCS heat removal performance
- Potential for stress corrosion cracking

The PSAR describes plans to address thermal fatigue in the RCCS that includes development of temperature fields for bounding transient and evaluation of thermal stresses using analysis consistent with ASME Section III, Division 5. Furthermore, the PSAR states that the fouling effects and stress corrosion cracking will be addressed using a water chemistry control program that maintains acceptable water quality and low chloride levels. The staff finds these proposed approaches to address the above design issues acceptable at the CP stage.

Based on the above, the NRC staff finds that the PSAR demonstrates that the RCCS design meets PDC 4.

Sharing of structures, systems, and components (PDC 5)

The NRC staff reviewed PSAR section 6.4.11.1.2, which states that none of the SR components of the RCCS (i.e., SFA subsystem) are shared between modules. Each module's SR SFA subsystem connects to a single, common NSRST PFA subsystem through dedicated cooling-water supply and return piping. The PFA subsystem consists of three cooling water pumps and six air-cooled heat exchangers shared by all four modules.

A key design feature supporting PDC 5 compliance is the elevation of the supply and return piping between the SFA and PFA subsystems. These lines are located above the nominal water level in each module's boil-off tanks. This configuration prevents loss of SFA inventory if a break occurs in the NSRST portion of the piping. When a break occurs, water levels in the RCCS loop fall, and once the level drops below the piping elevation, siphoning stops. This elevated geometry maintains the entire 72-hour boil-off inventory, even if PFA pumps continue running. As inventory decreases in the RCCS supply and return piping, the PFA pumps lose suction, which stops cross-module inventory loss and preserves SFA inventory in the unaffected modules. Because the supply and return piping elevation isolates 72-hour boil-off inventory, each module's SR SFA subsystem retains its capability to perform its safety function consistent with PDC 5.

In addition to this geometric design feature, the NSRST PFA subsystem includes motor-operated isolation valves on the supply and return piping. These valves are classified as NSRST components and they have no automatic closure function; instead, operators are expected to close them from the control room if a decreasing boil-off tank water level is observed. If there is break in the SFA subsystem of one module (the worst case SFA break is identified at the bottom of standpipe), the operators can close the isolation valves for the affected module to limit the inventory loss from the PFA subsystem. This would allow continued cooling of the unaffected modules. As described earlier, operator failure to isolate the affected module's SFA would not affect the 72-hour boil-off inventory of the unaffected modules due to the loss of suction for PFA pumps as soon as the RCCS loop water level drops below the top elevation of the supply and return piping. If a single module enters an AOO, DBE, or DBA without RCCS inventory loss, the PFA subsystem continues to cool the unaffected modules without impairment. The PFA subsystem provides 6 MWth of heat-removal capacity—roughly three times higher than the normal combined heat-removal needed for the four modules—so the shared NSRST components maintain adequate capability even during transients. Because operation of the isolation valves is not required to preserve the required safety function of the SR SFA subsystem, their NSRST classification and the associated operator action are acceptable. The PSAR notes that additional design features, such as automatic closure on low tank level, are being evaluated for the final design.

The PSAR notes that the structural foundation supporting RCCS SR and NSRST components, along with other safety-significant SSCs, is shared across modules. SE section 6.4.1 concludes that this shared foundation does not compromise compliance with PDC 5.

The RCCS maintains independent SR SFA subsystems, prevents cross-module inventory loss through elevated piping, ensures adequate cooling margin in the shared PFA subsystem, and provides geometric features that preserve required safety functions during AOOs, DBEs, and DBAs in any individual module. Based on the information above, NRC staff finds the RCCS design meets PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR section 6.4.11.1.2, which states that periodic inspections of SR RCCS SSCs will be conducted under the RIM program, which is discussed in section 8.2 of this SE. NRC staff reviewed the identified potential MANDE requirements, including visual examination, volumetric weld examinations, and secondary-water side chemistry monitoring. Because the design provides access and proposed monitoring and inspection addresses credible degradation mechanisms, the staff finds that the preliminary inspection and monitoring measures provide reasonable assurance that the functional capability of the RCCS SR SSCs will be maintained.

PSAR section 6.4.11.1.2 states that periodic inspections and functional testing for the NSRST RCCS SSCs will be performed in accordance with the RAM program described in PSAR section 8.10. Similar to the SR RCCS SSCs, the DMA process is used to assess the risk of degradation mechanisms and to select appropriate MANDE requirements for NSRST RCCS SSCs. Potential MANDE requirements for the NSRST RCCS SSCs include visual inspection, online leakage detection, and water chemistry monitoring. Functional testing for the active NSRST components will be performed in accordance with the maintenance program and include exercise testing and diagnostic testing. Because the design provides access, and proposed monitoring, inspection, and testing address credible degradation mechanisms, the NRC staff finds preliminary periodic inspection and functional testing of NSRST RCCS SSCs provide reasonable assurance that the functional capability of the RCCS will be maintained.

PSAR section 6.4.11.1.2 states that functional testing of RCCS will verify the operability and performance of its safety-significant components under conditions as close as practical to those expected during AOOs, DBEs, and DBAs. The planned functional testing is expected to encompass the conditions and ranges associated with operation of the SR RCCS SSCs. The adequacy of the test plan, including test conditions, measurements, and monitoring, will be evaluated during the OL review.

The PSAR also notes that RCCS monitoring indications for the SR and NSRST RCCS SSCs will be provided to operators through the DCS and PEMS for online monitoring. The adequacy of RCCS instrumentation and control design is addressed under the evaluation for PDC-CDC 13.

Based on the discussion above, the NRC staff find that the SR SSCs meets the requirements of PDC 6. The NRC staff will evaluate the RAM and maintenance programs as part of the OL application review, including the basis and acceptability of the departure from the scope of ASME Section XI-2 as endorsed in RG 1.246 (SE section 8.2) and OM-2 as endorsed in RG 1.220 (SE Section 8.4) which includes SSCs that perform functions requiring special treatment for defense in depth adequacy

Reactor Design (PDC 10)

The NRC staff reviewed PSAR section 6.4.11, which states that PDC 10 is a higher-level requirement that is met for the RCCS via application of PDC-CDC 34, which is aligned with performance of NSRST PSFs 1.2.2 and 1.2.2.3 to ensure that SARRDLs are not exceeded during AOOs. The PSAR further indicates that appropriate design margins that preclude

exceeding SARRDLs is inherent in the appropriately designed safety, quality, and seismic classification of RCCS, the application of appropriate codes and standards commensurate with the RSFs and PSFs for the RCCS, and the overall selection of materials with consideration of operating temperatures and pressures that the RCCS components would experience during normal operating conditions or AOOs. The NRC staff notes that SARRDLs were not included in the CP application, but the NRC staff reviewed the PSAR dose consequence results for AOOs for a preliminary assessment, considering margin to the F-C target.

The NRC staff finds the discussion is consistent with the NRC staff-approved Xe-100 PDC LTR, Rev. 3 and the updated basis for PDC 10 discussed in PSAR section 5.3. As discussed in the basis for PDC 10, it is a design limit that supports acceptance criteria in PDC-CDC 34. PDC-CDC 34 is aligned with the performance of NSRST PSFs 1.2.2 and 1.2.2.3 to ensure the SARRDL is not exceeded and that safe shutdown is achieved and maintained during AOOs. Therefore, PDC 10 requirements are satisfied by meeting the requirements in PDC-CDC 34. The evaluation of PDC-CDC 34 for the RCCS is presented later in this SE section.

PDC-CDC 13

PSAR section 6.4.11.2 describes how the RCCS instrumentation continuously measures key system parameters, including temperature, fluid level, pressure, and flow rate, and transmits these variables to the Main Control Room (MCR) through the Distributed Control System (DCS). For the PFA, the instrumentation measures SFA inlet and outlet temperatures and flow rate, enabling determination of the heat removal rate and providing assurance that the RCCS can remove sufficient heat to maintain temperatures within acceptable limits during AOOs.

The PSAR also states that boil-off tank levels are continuously monitored by both the DCS and the Post Event Monitoring System (PEMS), allowing operators to observe RCCS performance during DBEs and BDBEs. Together, the DCS and PEMS provide comprehensive monitoring capabilities that support operator awareness of RCCS functional status under all expected operating conditions.

Based on this information, the staff concludes that the proposed instrumentation is adequate to monitor the performance of the NSRST RCCS and meets the requirements of PDC-CDC 13.

Residual Heat Removal (PDC-RFDC 34)

PDC-RFDC 34 requires passive residual heat removal to ensure that fuel and radionuclide release limits are not exceeded during DBEs and DBAs. Consistent with this requirement, the system level SRDC for RCCS further requires that sufficient heat removal from the reactor pressure vessel (RPV) be maintained to limit RPV temperatures such that core geometry is preserved under DBE and DBA conditions.

The NRC staff reviewed PSAR section 6.4.11, which describes that the SR portion of the RCCS (i.e., SFA) serves as the passive boil-off residual heat removal system for DBAs and DBEs. The SR RCCS design includes two independent trains providing a combined 72-hour water inventory for passive boil-off cooling, with provisions for tank refill if extended passive cooling is required. As stated in the PSAR, the passive RCCS performance does not rely on operator actions, electric power, electric support systems, or electric interfaces to perform its RSF.

Preliminary analyses documented in PSAR sections 3.4 and 3.6 demonstrate the capability of SR RCCS to remove the decay heat under DBE and DBA conditions. Passive RCCS cooling is credited in multiple DBE scenarios evaluated in PSAR section 3.4. The analyses in sections 3.4.7 and 3.4.8 (non-DBA PLOFC and DLOFC) show decreasing average and peak core temperatures during the long-term phase when passive RCCS heat removal is available. PSAR tables 3.2-3 and 3.2-4 show that 30-day DBE doses remain below applicable acceptance criteria. PSAR sections 3.6.10 and 3.6.11 provide corresponding results for DBA PLOFC and DLOFC scenarios, representative of long-term phases of multiple DBAs and PSAR table 3.2-5 confirms that 30-day doses at the EAB following DBAs remain below regulatory limits.

Although the PSAR does not provide corresponding RPV wall and core metallic structure temperatures for these scenarios, PSAR section 3.4.1 presents RPV temperature predictions for the MLOOP DBE using Flownex in PSAR figure 3.4.1-2. These results indicate that the long-term RPV outer surface temperature increases to approximately 500°C, with average fuel temperatures of approximately 800°C. These preliminary results suggest that the RCCS limits RPV structural temperatures below applicable ASME III Division 5 design criteria, consistent with PDC-RFDC-34 and the SRDC for RCCS.

The PSAR notes in section 3.1.3 that the GOTHIC–Flownex evaluation models used for the preliminary analyses are based on an air-cooled RCCS configuration, whereas the design described in section 6.4.11 employs a water-cooled, boil-off RCCS. The PSAR states that the air-cooled RCCS modeling provides conservative results because the water-cooled RCCS is expected to provide more effective heat removal through boil-off heat transfer and enhanced surface area and heat transfer coefficients. Verification of this assumption, including confirmation that the water-cooled RCCS can perform its required safety functions, will be provided at the OL application stage through updated models and additional DBE/DBA analyses. Detailed evaluation of the RCCS modeling approach is addressed in the portion of this SE titled “Preliminary Analysis of RCCS” below, including the NRC staff’s evaluation of the approach’s compliance with 10 CFR 50.34(a)(4) and 10 CFR 50.35(a).

During the regulatory audit [MLxxx], the staff identified several RCCS design issues that could affect long-term passive cooling performance. The PSAR identifies and commits to resolution of these issues at the final design stage:

1. Separation of steam in the riser standpipes and headers and its venting to atmosphere at sufficient rate is important for effective performance of RCCS during the boil-off phase. The PSAR states that the effectiveness of RCCS design to achieve gravity-driven separation of vapor and liquid (or steam and droplets) in the riser header will be evaluated during the final design phase.
2. Boiling instabilities such as geysering and parallel channel instabilities could induce mechanical and thermal stresses on the RCCS structures and affect the RCCS performance. The PSAR commits to evaluating these phenomena at the OL application stage and implementing design changes as necessary. The PSAR further commits to evaluating thermal stresses associated with various RCCS operating modes and event scenarios during final design.
3. Level swell in riser pipes and headers may result in entrainment and loss of RCCS inventory through vent lines. The PSAR states that the assumption that vapor generation

rates are slower than vapor–liquid separation rates will be evaluated at the final design stage, and design modifications will be implemented as needed.

4. Physical processes governing the transition from forced to natural circulation and the onset of nucleate boiling may significantly influence performance. The PSAR commits to evaluating the assumptions regarding natural circulation initiation and vapor–liquid separation rates during final design.
5. Two-phase flow and non-uniform thermal conditions along riser standpipes may challenge RCCS structural integrity. The PSAR states that low power density and natural circulation are expected to limit temperature variation, and that standpipe bowing can be mitigated through structural supports. The adequacy of these assumptions and the corresponding support design will be evaluated with additional analysis during final design.

As evaluated in portion of this SE below titled "Additional Testing, Analyses, and Design Development," the commitments to address these design issues through additional analyses and potential design improvements at the OL stage provide the NRC staff with reasonable assurance that the required information will be supplied in the FSAR and therefore support issuance of a CP in accordance with 10 CFR 50.35(a).

The NRC staff's evaluation of the applicant's plan for pre-operational testing of the as-built RCCS design is discussed later in this SE under the FOAK testing subsection, consistent with treatment of test program requirements in 10 CFR Part 50, Appendix B, Criterion XI.

Based on the description of the RCCS and the preliminary analyses for DBEs and DBAs performed using a conservative RCCS model and demonstrating that the RCCS can maintain fuel and RPV temperatures below applicable design limits, the NRC staff concludes that the preliminary RCCS design is consistent with PDC□RFDC□34. However, the NRC staff notes that several RCCS design issues must be resolved through further analysis or design modifications and will be reviewed at the OL application stage. Furthermore, the NRC staff notes that a validated RCCS evaluation model, benchmarked against applicable experimental data, will be required at the OL application stage to demonstrate compliance of the final RCCS design with PDC□RFDC□34.

PDC CDC 34 - Residual Heat Removal

PDC CDC 34 requires an active means to transfer fission product decay heat and other residual heat from the reactor core to an ultimate heat sink at a rate such that specified acceptable system radionuclide release design limits are not exceeded during AOOs.

As described in PSAR section 6.4.11, the NSRST portion of the RCCS (i.e., PFA) serves as an active heat removal system during normal operation and certain LBEs. The PFA is integrated with the SR RCCS (i.e., SFA) trains through supply and return piping and includes pumps, heat exchangers, and instrumentation.

The preliminary analyses in PSAR section 3.3 through 3.5 demonstrate RCCS capability to remove decay and residual heat during LBEs. For example, PSAR table 3.2-2 shows that doses for AOO event sequences SSGTL-03 and STLPF-07, which credit active RCCS cooling, remain below acceptance criteria. Additionally, PSAR table 6.4.11-2 provides process parameters for

NSRST RCCS during normal operation, confirming low operating temperatures **[[ ]]**. The PSAR also states that total heat removal capacity of the PFA subsystem is approximately 6 MWth. This is three times the total heat load from all four modules during normal operation and AOOs.

Based on these analyses, the staff concludes that the preliminary NSRST RCCS design information meets the requirements of PDC CDC 34.

Preliminary Analysis of RCCS

10 CFR 50.34(a)(4) requires a preliminary analysis and evaluation of the design and performance of SSCs of the facility. As discussed earlier in this section, PSAR sections 3.3 through 3.6 present the preliminary analyses for selected AOOs, DBEs, and DBAs that demonstrate the effectiveness of the RCCS in providing active and passive cooling to prevent or mitigate these events. These analyses demonstrate that the preliminary RCCS design can maintain acceptable fuel and RPV temperatures under the evaluated conditions.

However, the staff notes that the PSAR does not clearly demonstrate the fidelity of the RCCS evaluation model used in these analyses relative to the water-cooled RCCS design described in PSAR section 6.4.11. The GOTHIC-Flownex EM used to generate the PSAR preliminary results was developed following the evaluation model development and assessment process (EMDAP) guidance in RG 1.203. However, as discussed earlier in this SE, the preliminary RCCS model in GOTHIC-Flownex is based on air-cooled RCCS. Nevertheless, as discussed earlier in this SE, the RCCS EM employed in the PSAR is based on an air-cooled RCCS configuration and does not yet incorporate key thermal-hydraulic phenomena associated with the boil-off, water-cooled RCCS.

PSAR section 6.4.11.3 states that qualification and validation of the GOTHIC-Flownex RCCS EM are being performed using integral effects test (IET) data from the Argonne National Laboratory (ANL) Natural Convection Shutdown Heat Removal Test Facility (NSTF). The applicant notes that NSTF is an NQA-1 facility and that its test database covers a broad range of operating conditions, power levels, and thermal-hydraulic behaviors relevant to passive RCCS heat removal. While this provides a reasonable preliminary basis at the CP stage, the staff finds that a systematic scaling and distortion analysis is needed to establish the applicability of NSTF data to the Xe-100 RCCS across the full range of conditions expected during long-term passive cooling. The applicant indicates that such an applicability assessment is underway and will include both global (top-down) and component-level (bottom-up) evaluations. The PSAR further states that any gaps identified through scaling analysis will be addressed through supplemental analyses and, where necessary, design modifications. If the applicant determines that NSTF data are insufficient to fully validate the RCCS EM, the PSAR indicates that additional testing will be performed under an appropriate test program.

Although the current RCCS EM does not adequately represent the water-cooled RCCS design, the staff finds that the use of a conservative, air-cooled RCCS model is acceptable for the CP stage because preliminary modeling approaches are permissible at the CP stage in accordance with 10 CFR 50.34(a)(4) and the applicant identifies the remaining technical work that can be reasonably left for the OL application stage in accordance with 10 CFR 50.35(a)(2). The applicant has clearly identified the scope of remaining technical work necessary to produce a

water-cooled RCCS evaluation model consistent with RG 1.203, including scaling and distortion analyses, experimental validation, and potential additional testing. These commitments provide reasonable assurance that an applicable and validated RCCS EM will be available to support the OL application safety analyses.

Technical Specifications for RCCS (Compliance with 10 CFR 50.34(a)(5))

10 CFR 50.34(a)(5) requires identification and justification of variables, conditions, or items that, based on preliminary safety analysis, are likely to become subjects of technical specifications for the facility. CPA Part IV, Table IV-3 identifies preliminary systems subject to Limiting Conditions for Operation (LCOs), including the requirement to maintain sufficient water inventory in the RCCS boil-off tanks to ensure that the safety-related portion of the RCCS can perform its required safety functions. The staff finds that identification of this LCO is adequate given that the passive SR RCCS does not depend on electric power or operator action for its operation. Based on this information the staff determines that the RCCS design described in the PSAR satisfies the requirements of 10 CFR 50.34(a)(5). Consistent with 10 CFR 50.34(b)(6)(vi), the NRC staff will evaluate the proposed technical specifications for RCCS when detailed design information is available at the OL application stage.

Additional Testing, Analyses, and Design Development

PSAR section 6.4.11.2 states that the supply and return piping are located above the water tank level to prevent siphoning in the event of a piping break. This design implies that, under normal operation, all passive portions of the RCCS—including tanks, downcomer and riser headers, and associated standpipes—are filled with subcooled water. However, the PSAR does not clearly describe the heat removal governing transition from active cooling to passive boil-off cooling following loss of forced circulation. Although the fundamental phenomena and processes at the subsystem and component level for the RCCS are generally well understood (e.g., radiation, conduction, convention, and boiling heat transfer), the unique geometry and configuration of passive RCCS introduces significant uncertainty regarding integral system performance over time, particularly during the long-term cooling phase. Several critical aspects of passive RCCS performance remain insufficiently characterized and may require additional testing, analyses and development to validate design basis. These include:

- Mechanisms for steam separation and discharge from the riser standpipe and riser header to the atmosphere.
- Potential for boiling instabilities (e.g., geysering, parallel channel instabilities) and their impact on passive RCCS performance.
- Risk of level swell in the riser and riser header and associated entrainment of RCCS inventory through riser vents.
- The transition from active to passive cooling and two-phase flow behavior in the risers.

In addition, the validation basis for analytical models and associated tools used to predict integral RCCS performance, including scaling analysis, applicability evaluations, and closure-model development are necessary to confirm RCCS performance and have not yet been completed.

Although the ANL NSTF facility is the closest geometric and phenomenological analog to the proposed RCCS, significant differences remain between NSTF and RCCS geometries. Examples include:

- Closed-loop NSTF configuration vs. open loop RCCS
- Cooling panels in NSTF vs. downcomer/riser standpipes in RCCS
- Small diameter of NSTF cooling panel tubes vs. large diameter standpipes in RCCS
    - Fundamentally different two-phase flow regimes between small and large diameter pipes
    - Different characteristics of two-phase flow instabilities between small and large diameter pipes
- NSTF Headers vs. RCCS downcomer and riser headers

These geometric and phenomenological differences introduce significant uncertainties in the applicability of NSTF data to the proposed RCCS design.

As noted in the evaluation of PDC-RFDC 34 above, the applicant identified steam separation, boiling instabilities, level swell, the transition from active to passive cooling, and two-phase flow behavior as RCCS design issues and has committed to resolving these issues through additional analysis or design modifications for the final design.

Additionally, the PSAR outlines the development activities required to produce a water-cooled RCCS evaluation model consistent with RG 1.203, including top-down and bottom-up scaling and distortion analyses, experimental validation using NQA-1 NSTF data, and if needed additional integral-effects or separate-effects testing. Furthermore, the LMGS R&D program evaluated in Chapter 13 indicates that the SSCs like RCCS may be incorporated into the R&D program if testing or analysis reveals safety questions requiring resolution to confirm the system's adequacy. The commitments described above provide the NRC staff with reasonable assurance that the required information, analyses, and evaluations described above will be supplied in the FSAR and therefore support issuance of a CP in accordance with 10 CFR 50.35(a).

FOAK Testing

Criterion XI, "Test Control," of Appendix B to 10 CFR Part 50 states, in part, that, "[a] test program shall be established to assure that all testing required to demonstrate that SSCs will perform satisfactorily in service is identified and performed in accordance with written test procedures, which incorporate the requirements and acceptance limits contained in applicable design documents." PSAR section 6.4.11.3 states that no RCCS components have been identified as incorporating FOAK technology, design features, or configurations warranting FOAK special treatment testing to demonstrate that the RCCS can perform its SR RSFs and NSRST PSFs.

The NRC staff finds that this assessment is reasonable for the NSRST portion of the RCCS (the PFA subsystem), which employs conventional industry equipment such as pumps, piping, and

heat exchangers and operates under forced-circulation conditions. As a result, preoperational testing identified under PDC 6 is considered adequate for the PFA subsystem.

However, for the SR RCCS (the passive SFA subsystem), the NRC staff notes that although the underlying thermal-hydraulic phenomena are well understood at the component level (e.g., radiation, conduction, convection, boiling heat transfer), the integral geometry and configuration of the passive SFA are unique, and no prior reactor design or test facility has implemented an RCCS configuration directly comparable to that proposed by the applicant. This unique passive arrangement, coupled with the safety significance of long-term decay heat removal, necessitates assurance that the as-built passive RCCS will perform as intended before it is relied upon during plant operation or integral plant testing.

As discussed earlier in this SE and reflected in the PSAR, the applicant plans to use experimental data from the ANL NSTF to demonstrate the performance of the passive RCCS and to qualify and validate the GOTHIC-Flownex evaluation model. The applicant has stated that NSTF is an NQA-1 facility and that its data envelope spans a wide range of operating conditions and phenomena relevant to RCCS behavior. The applicant further commits to performing scaling and distortion analyses to establish the applicability of NSTF data to the Xe-100 RCCS design. Should these analyses identify gaps that cannot be resolved through additional evaluation or design modification, the applicant commits to conducting additional testing under an appropriate test program.

Additionally, in demonstrating compliance with PDC 6, the applicant states that pre-operational and startup testing will be used to confirm RCCS operability and performance under conditions as close as practical to those expected for AOOs, DBEs, and DBAs. While the applicant does not classify the SR RCCS as a FOAK system, the NRC staff determines that the applicant's combined plan, which includes NSTF data analysis, global and component-level scaling and distortion evaluation, potential design modifications and supplemental testing if needed, and pre-operational testing under conditions approximating accident scenarios, provides adequate assurance that the performance of the as-built RCCS will be demonstrated prior to its use in operation or integral plant testing. Based on this evaluation, the NRC staff concludes that LME's determination that the SR RCCS components and design features do not warrant FOAK special treatment is reasonable. The NRC staff will evaluate the continued validity of the FOAK treatment at the OL application stage as the RCCS design matures and detailed design information is available.

Safety Functions

Control Heat Removal Function

PSAR section 6.4.11.1 indicates that the SR portion of the RCCS is designed to meet the following RSFs:

RSF 1.2.1, "Maintain core heat removal through passive means"
RSF 1.2.1.3, "Transfer heat from reactor cavity to the ultimate heat sink"

The RCCS applies the plant-level PDC-RFDC 34, "Residual heat removal," to support performance of RSFs 1.2.1 and 1.2.1.3.

The following system-level SRDC is applied to the RCCS to ensure successful performance of RSFs 1.2.1 and 1.2.1.3:

> The RCCS is designed to (1) transfer sufficient heat from the reactor cavity to the ultimate heat sink to ensure that fuel and radionuclide release limits are not exceeded during DBEs or DBAs and (2) ensure sufficient heat transfer from the RPV to limit the RPV temperature to maintain core geometry during DBEs and DBAs.

The RCCS is also designed to meet the following PSF, which is required to be performed during specific LBEs:

> PSF 1.2.2, "Maintain core heat removal through active means"
> PSF 1.2.2.3, "Control heat removal of the reactor cavity through active means"

The PSFs 1.2.2 and 1.2.2.3 support adequate DID for RSF 1.2. The RCCS applies the plant-level PDC-CDC 34, "Residual heat removal," and the following system-level CDC to support performance of PSFs 1.2.2 and 1.2.2.3:

> The RCCS is designed to transfer sufficient heat from the reactor cavity to the ultimate heat sink to ensure that specified acceptable system radionuclide release design limits are not exceeded during AOOs.

The RCCS is designed to perform RSFs 1.2.1 and 1.2.1.3 with no reliance on operator action or electrical power. Electrical power is required for the RCCS to support the PSFs 1.2.2 and 1.2.2.3. As such, the staff determined that the preliminary RCCS design supports the control heat removal function.

The NRC staff finds the PSAR description of the RCCS to be consistent with the guidance for the content of CP applications found in section C.6.4, "Descriptions for SR SSCs," of NEI 21-07 as endorsed by NRC RG 1.253.

6.4.11.2 Conclusion

The NRC staff concludes that the preliminary RCCS design is acceptable and meets the requirements of 10 CFR 50.34(a)(3)(i), 10 CFR 50.34(a)(3)(ii), 10 CFR 50.34(a)(3)(iii), 10 CFR 50.34(a)(4), and 10 CFR 50.34(a)(5). This conclusion is based on the following:

- The applicant identified the applicable PDCs applicable to the RCCS and described the preliminary design of the RCSS.
- The applicant described how the RCCS meets the PDCs applicable to the RCCS.
- The applicant provided the materials of construction, approximate dimensions and general arrangement of the RCSS.
- The applicant presented the results of preliminary analysis of effectiveness of RCCS in prevention or mitigation of AOOs, DBEs, DBAs.
- The applicant identified key RCCS variables and conditions as probable subjects of technical specifications.

The NRC staff recognizes the preliminary nature of the RCCS design as well as the maturity of the PRA used to identify the RSFs and PSFs to meet the required functions of the RCCS. The applicant identified in the PSAR the gaps in the preliminary RCCS design and associated evaluation model and identified its plans for addressing those gaps in the final design to be submitted as part of an OL application. As a result, the NRC staff finds, in accordance with 10 CFR 50.35(a), that further technical or design information required to complete the evaluation, as part of the NEI 18-04 process implementation, and which can reasonably be left for consideration at the OL application stage, will be supplied in the FSAR. The NRC staff will confirm that the RCCS systems conform to the design basis during the evaluation of the FSAR as part of the OL application review.

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6.4.12 Steam Generator System

PSAR section 6.4.12 describes the steam generator system (SGS). The primary function of the SGS is to transfer heat from the primary helium to the water/steam cycle. The SGS also accepts condensate and provides steam to the startup and shutdown system (SSS) during certain LBEs. The SGS consists of the steam generator (SG), steam generator feedwater system (SGFS), and the steam generator steam system (SGSS).

System specific regulatory guidance applicable to the SGS is as follows:

- Regulatory Guide 1.87, Revision 3, “Acceptability of ASME Code Section III, Division 5, 'High Temperature Reactors,'” (ML25176A084).
- RG 1.26, Rev. 5, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants Staff Regulatory Guidance 1,” (ML16082A501).
- Regulatory Guide 1.29, Revision 6, “Seismic Design Classification for Nuclear Power Plants,” (ML21155A003).

In addition to PDCs 1 through 6, the following system specific PDCs are applicable to the SGS:

- PDC 10 - Reactor Design
- PDC-CDC 13 - Instrumentation and Control
- PDC 14 - Reactor Helium Pressure Boundary
- PDC-CDC 15 - Reactor Helium Pressure Boundary Design
- PDC-CDC 16 – Functional Containment Design
- PDC-RFDC 30 and PDC-CDC 30 – Integrity of Reactor Helium Pressure Boundary
- PDC 31 - Fracture Prevention of Reactor Helium Pressure Boundary
- PDC-CDC 34 - Residual Heat Removal

6.4.12.1 Technical Evaluation

Safety Significance

The SGS has both SR and NSRST SSCs, with the SR SSCs being the nuclear island feedwater isolation (NIFI) valves, the nuclear island steam isolation (NISI) valves, and the secondary relief valves (SRVs). The function of these valves is to limit water/steam ingress into the HPB to acceptable levels in the event of a tube rupture to minimize oxidation of graphite and metallic SSCs, which would challenge the functions and structural integrity of SSCs susceptible to oxidation. Risk-significant NSRST isolation valves between the HSS and SGS isolate the primary helium side of the SG from a depressurization event initiated in the HSS. The SGS also supports DID functions for maintaining active heat removal so that the fuel in the core stays below the temperature limits and maintains integrity as the primary radionuclide retention barrier.

The SGS also has NSRST SSCs, which make up portions of the HPB. These include the SG pressure vessel assembly, the tubes and tubesheets, the piping and isolation valves to the HSS, and the moisture detection piping and isolation valves. The primary safety-significant function of these NSRST SSCs is maintaining the HPB integrity to prevent moisture ingress. The NRC staff's review of the preliminary SGS design focused on design areas that are outside of current Codes and Standards, such as the temperature and lifetime limits in ASME Section III, Division 5, design of welds that fall outside of ASME III.5, and environmental compatibility.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR section 6.4.12, which states that the SGS comprises both SR and NSRST SSCs. The SR SSCs consist of the NIFI, NISI and SRV valves. Per table 6.4.12-2 these components will be designed to ASME BPVC Section III, Division 5, Subsection HC, Subpart A or B, which is endorsed in RG 1.87, Rev 3. Subsection HC is an acceptable subsection for construction of the SR components of the SGS because it is intended for Class B metallic pressure boundary components. ASME III.5.HCB states that the rules do not explicitly address fatigue damage from cyclic service. LME acknowledged via audit that, if necessary in the final design, fatigue will be appropriately accounted for, potentially by incorporation of fatigue rules from HBB. The materials of construction listed in PSAR table 6.4.12-2 will meet the design and normal service level conditions in accordance with ASME BPVC III.5.HC. LME performed construction code assessment using the methodology, as described in PSAR section 3.8, and determined that no supplementation is necessary for the SR SSCs based on the preliminary design. Based on the information above, the NRC staff finds that the preliminary design of the SR SGS meets the requirements of PDC 1 to use Codes and Standards commensurate with the safety significance of the component safety functions.

The NRC staff reviewed the NSRST SSCs, which are identified in table 6.4.12-3, as well as the service conditions, materials, and construction codes. The metallic NSRST components of the SGS will be designed to ASME BPVC Section III, Division 5, Subsections HB, HC, HF, and HG, subparts A and B, as applicable, for the specific SSCs. ASME III.5 is endorsed in RG 1.87, Rev.

3, and is a preliminary special treatment for NSRST SSCs. LME performed construction code assessment using the methodology as described in PSAR section 3.8. The NRC staff reviewed PSAR table 6.4.12-5, which contains the results of this assessment for the SGS and the applicant's plans to address the areas needing supplementation with the materials qualification plan. The areas requiring supplementation of the ASME III.5 rules include: [[ ]]. In addition, ASME BPVC III-5 does not address the design of several types of welds that are being used in the SGS, which include dissimilar metal (DM) welds, tube-to-tubesheet welds, and tube-to-tube welds, and these areas are included in the materials qualification plan. Assessments for alloy 800H that overlap with the FUCH are included in the R&D program, which is reviewed in Chapter 13 of this SE. For the remaining items, the staff reviewed the applicant's plans to conduct assessments and testing under the materials qualification plan provided in PSAR table 6.4.12-5 and finds the plans adequate to support the preliminary design. In addition, LME stated in the PSAR where Code supplementation areas will be incorporated into ASME III.5 Code Cases, which would provide additional confidence in the adequacy of the materials qualification program. Based on the information above, the NRC staff finds that the preliminary design of the SGS system meets the requirements of PDC 1 to use appropriate Codes and Standards commensurate with the safety significance of the component safety functions.

The applicant applied ASTM C795, "Standard Specification for Thermal Insulation for Use in Contact with Austenitic Stainless Steel," to the NSRST internal insulation supports. The NRC staff finds ASTM C795 an acceptable standard for minimizing the potential for stress corrosion cracking of the underlying 316H stainless steel caused by leaching of chlorides from the insulation as it is an industry recognized consensus standard. The applicant did not identify a similar ASTM standard for ensuring the critical characteristic of the NSRST thermal insulation (i.e., thermal conductivity). However, common industry practice is to identify a standard for measuring the thermal conductivity of the insulation prior to procurement, and doing so at the CP stage is not required. As such, the NRC staff will review the standard verifying the thermal conductivity of the NSRST insulation at the OL application stage.

The full scope of the LMGS QAPD, described in PSAR section 8.1, is applied to the SR SGS SSCs. This program is designed to meet 10 CFR Part 50, Appendix B, and is appropriate for meeting the quality assurance requirements under PDC 1 for the SR aspects of the SR SGS SSCs. The NSRST SGS SSCs are designed, fabricated, constructed, and tested in accordance with the quality assurance program described in Part III of the LMGS QAPD and in PSAR section 8.1. Additional special treatments related to quality assurance will be specified at the OL application stage. Application of Part III of the LMGS QAPD at the CP stage is sufficient as a preliminary special treatment to meet the quality assurance requirements under PDC 1 for the NSRST portions of the SGS. This finding is based on the NRC staff's evaluation of the LMGS QAPD in the QAPD TR SE and in section 8.1 of this SE.

Based on the review described above, the NRC staff determined that the preliminary SGS design is consistent with PDC 1 because the Codes and Standards specified for the design and construction are acceptable and, where supplementation is necessary, adequate information on the materials qualification plan is provided.

Design Basis for Protection against Natural Phenomena (PDC 2)

The NRC staff reviewed PSAR section 6.4.12, which states that the SGS SSCs located within the reactor building are protected from applicable natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions by their location. PSAR Section 6.4.12 also states that some NSRST SSCs are located in the NIAS and the SST, or between the reactor building and NIAS, and these SSCs will be designed to resist DHBLs described in PSAR section 6.1.1. Additionally, PSAR section 6.4.12 states that the location of these NSRST SSCs will be designed to ensure that missiles created by natural phenomena will not compromise the safety function of these NSRST SSCs. LME states that the missile protection design basis, analysis methodology, and design features will be provided for the staff to review at the OL stage.

The SGS is designed to withstand seismic events and achieve performance levels corresponding to its designated seismic category as shown in PSAR table 3.7-2 and discussed in PSAR section 3.7. As stated in PSAR table 3.7-2, the SR portions of the SGS are categorized as seismic Category I to ensure they can perform the associated RSFs during and following a seismic event. In addition, SSCs that could prevent the SGS from performing their RSF due to spatial interaction (i.e., seismic II/I) are categorized as seismic Category IIB to prevent this interaction.

Per PSAR section 6.1.3, additional hazard protection strategies may include physical separation where feasible, protection barriers, or designing the SR SSCs to withstand impact from SSC failures. The final strategies used and associated processes and methods will be provided at the OL application stage.

Some NSRST components of the SGS are categorized as seismic Category IIA for preserving dynamic characteristics to ensure they can provide adequate support to SR SSCs during and following a seismic event. Other NSRST components of the SGS are categorized as seismic Category IIA for seismic safety significance to ensure they can perform the associated NSRST PSFs during and following a seismic event. These include the SGS NSRST components relied upon for maintaining HPB integrity to prevent moisture ingress.

The seismic design requirements associated with each category and the codes and standards applied are described in PSAR section 3.7 and evaluated in section 3.7 of this SE. The NRC staff finds this seismic categorization and the associated design approach to be sufficient to address seismic aspects of PDC 2 for the preliminary design stage. Additional information on the final seismic design and the SSCs identified as needing seismic category IIB categorization to prevent interaction with this system or other strategies to protect the system from adverse interaction will be reviewed at the OL application stage as part of the final design.

The PSAR states that hazard protection strategies, processes, and methods which are part of design considerations will incorporate more detailed deterministic and probabilistic hazard assessments at the OL application. .

Based on its review, as discussed above, the NRC staff finds that the preliminary SGS design features and the associated design development approach described in the PSAR are consistent with PDC 2. The NRC staff will review the detailed deterministic and probabilistic hazard assessments at the OL application stage to determine whether the final SGS design meets the requirements of PDC 2

Fire Protection (PDC 3)

The metallic portions of the SGS are made of high-temperature-resistant and inherently non-combustible metallic materials. Likewise, the insulation used in the SGS is made of high-temperature-resistant and inherently non-combustible ceramic material. The SGS is located in the steam generator citadel where fixed combustible materials are negligible and transient combustibles are administratively controlled. SR SGS SSCs are located in the SST. PDC 3 for the SST is reviewed by the staff in section 6.4.1 of the SE. In addition, PSAR section 6.4.12 states that the SGS is protected by fire protection systems and a comprehensive plant fire protection plan (FPP). These fire protection systems and the FPP are developed and maintained to meet the requirements of 10 CFR 50.48(a)(1) and PDC 3 and are evaluated in section 8.6 of this SE. Based on the above information, the NRC staff finds that the SGS design meets the requirements of PDC 3.

Environmental and dynamic effects design bases (PDC 4)

The NRC staff reviewed PSAR section 6.4.12, in which LME states that PDC 4 requirements on the design to accommodate the effects of and to be compatible with the environmental conditions are satisfied by performing the environmental compatibility assessment and developing the material qualification plan to mitigate the applicable material degradation modes identified from the material compatibility assessment, as described in PSAR section 3.8. As discussed in the PDC 1 evaluation, PSAR table 6.4.12-5 identifies areas that are currently found necessary for the SGS that are outside of ASME Code rules and require supplementation. The NRC staff reviewed the preliminary assessment and planned testing program and finds that it will adequately identify applicable and non-applicable degradation modes, determine a strategy for mitigation of applicable degradation modes and identify operational programs, such as in-service monitoring and chemistry control. LME states that the SGS materials are not expected to experience significant corrosion in the helium environment. In addition, environmental qualification work tied to Alloy 800H, such as an assessment of irradiation effects of mechanical properties, is in the R&D program and would be available to support the SGS design if necessary.

The NRC staff also reviewed PSAR section 6.4.12 to assess whether it meets PDC 4 requirements of protecting the SGS system against dynamic effects. PSAR Section 6.4.12 identified potential sources of dynamic hazards to the SGS and stated that as the design matures, the credibility of these dynamic hazards will be systematically assessed and addressed. LME states that analyses will be performed to demonstrate adequate protection of the SGS, including evaluation of external pressure loads from rapid compartment pressurization, and assessment of impingement generated by feedwater, steam, or helium discharge. In addition, LME states that design measures to mitigate dynamic effects include incorporation of component supports and shielding as well as strategic equipment location. The NRC staff finds that the PDC 4 requirements of protecting the SGS system against dynamic effects are met for the preliminary design by committing to systematically assess and address the dynamic hazards identified. The NRC staff will review the final design approach and supporting analyses to demonstrate compliance with PDC 4 on protection against dynamic effects, which will be documented in the FSAR to be reviewed for the OL application.

The PSAR identifies risk-significant NSRST isolation valves that are credited with performing HPB isolation functions. The NRC staff evaluation of active equipment qualification for mechanical SSCs is found in SE section 3.8.

Based on the above discussion, the NRC staff finds that the preliminary design of the SGS system meets the requirements of PDC 4. Staff will evaluate the maturation of the R&D and materials qualification programs as well as the final design during the OL application review.

Sharing of structures, systems, and components (PDC 5)

The NRC staff reviewed PSAR section 6.4.12, which states that the SGS components are not shared between reactor modules. The PSAR also states that SGS components share a foundation system with multiple safety-significant SSCs. The adherence to PDC 5 within the context of a shared foundation system is described in PSAR section 3.7.6.3 and 6.4.1. The NRC staff evaluated this and found it acceptable in section 6.4.1 of this SE. Based on the discussion above, the NRC finds that the preliminary design of the SGS system meets the requirements of PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR section 6.4.12 for compliance of the SGS preliminary design with the requirements in PDC 6 for both SR and NSRST components.

For SR and risk significant NSRST components, LME states that periodic inspections of the SR passive SGS components are performed in accordance with the RIM program, which is evaluated in section 8.2 of this SE. The SGS design provides access for implementing remote preservice and inservice requirements. The NRC staff reviewed the identified potential MANDE requirements, including visual examination of the SR valve bodies and secondary side water chemistry monitoring. Because the design provides access and proposed monitoring and inspection addresses credible degradation mechanisms, the staff finds that the preliminary monitoring measures provide reasonable assurance that the functional capability of the SR SGS will be maintained. Functional testing of the SR active components will be performed in accordance with the IST program, which is evaluated in SE section 8.4. The NISI and NIFI valves are designed to be tested in service. The NRC staff reviewed potential IST requirements, which include exercise testing, and finds them appropriate to provide adequate assurance to maintain the functional capability of the NSRST active SSCs. In addition, LME states that the preoperational testing as described in PSAR section 12.1.1 will be performed to provide a baseline for the inservice inspections, which is evaluated in SE section 12.1.1.

As confirmed by NRC staff via a Request for Confirmatory Information (ML26239A157), LME is in the process of incorporating the results of new risk-significance evaluations following the NEI 18-04 methodology. Based on a preliminary evaluation, LME identified the SGS as risk-significant,, potentially impacting the scoping of NSRST SSCs in operational programs. As stated in PSAR sections 8.2 and 8.4, SSCs that are risk-significant will be included in the RIM and IST programs, and the NRC staff will evaluate the operational program scoping of SSCs in the SGS system during the OL application review based on final risk-significance determinations.

For NSRST components, LME states that periodic inspections of the NSRST passive components are performed in accordance with the RAM program, which is discussed in SE section 8.10. The design provides access for implementing remote preservice and inservice requirements. The NRC staff reviewed the identified potential MANDE requirements, including eddy current testing for preservice and inservice inspections as well as helium side and secondary-water side chemistry monitoring. Because the design provides access and proposed

monitoring and inspection addresses credible degradation mechanisms, the staff finds that the preliminary monitoring measures provide reasonable assurance that the functional capability of the NSRST SGS SSCs will be maintained. LME states that functional testing of the NSRST PPRS active components will be performed in accordance with the maintenance program, which is evaluated in SE section 8.3. The NRC staff reviewed potential IST requirements, including exercise testing and diagnostic testing, and finds them appropriate to provide adequate assurance to maintain the functional capability of the NSRST active SSCs. However, the maintenance program does not specify a method for developing and implementing an IST program, such as ASME OM-2. The NRC staff will review the final NSRST ISI and IST programs at the OL application. In addition, LME states that the preoperational testing, as described in PSAR section 12.1.1, will be performed to provide a baseline for the inservice inspections, which is evaluated in SE section 12.1.1.

Based on the discussion above, the NRC staff finds that the preliminary design of the SR SGS SSCs meets the requirements of PDC 6. For the NSRST SGS SSCs, the NRC staff will evaluate the RAM and maintenance programs as part of the OL application review, including the basis and acceptability of the departure from the scope of ASME Section XI-2 as endorsed in RG 1.246 (SE section 8.2) and OM-2 as endorsed in RG 1.220 (SE Section 8.4) which includes SSCs that perform functions requiring special treatment for defense in depth adequacy

Reactor Design (PDC 10)

The PSAR states that PDC 10 is a higher-level requirement for reactivity control that is met via application of PDC-CDC 34, “Residual Heat Removal.” The staff finds the SGS meets the requirements of PDC-CDC 34, as discussed in the evaluation of PDC-CDC 34 below.

Instrumentation and Control (PDC-RFDC and CDC 13)

The SGS meets the requirements of PDC-RFDC 13 by locating instrumentation in appropriate locations within the SG and piping systems to monitor key process parameters, e.g., helium moisture levels, pressure, temperature and flow rates. The instrumentation includes SR sensors for detecting moisture ingress. These sensors are used by the RPS to trip the NIFI and NISI valves, the helium circulators, and the reactor. These sensors are also used to indicate that primary flow can be restored to support associated RSFs and PSFs. The NRC staff evaluation of the RPS is in section 6.4.13 of this SE. The NRC staff finds the preliminary information discussed above to be consistent with the requirements of PDC-RFDC 13.

The SGS also meets the requirements of PDC-CDC 13 by locating instrumentation to monitor pressure, temperature, and other variables to support the functions of safety-significant SSCs, in particular identifying depressurization events to isolate the SGS from the HPS. The IPS, which is evaluated in SE Section 7.3.11, includes the logic circuits and interlocks to control automatic actions, when needed, to perform associated PSFs. The NRC staff finds the preliminary information discussed above to be consistent with the requirements of PDC-CDC 13.

PDC 14 – Reactor Helium Pressure Boundary, PDC CDC 15 – Reactor Helium Pressure Boundary Design, PDC CDC 16 – Functional Containment Design, & PDC 31 - Fracture Prevention of Reactor Helium Pressure Boundary

The NRC staff reviewed the PSAR section 6.4.12 application of PDC 14, PDC-CDC 15, PDC-CDC 16, and PDC 31 to the following systems 1) the SGPV assembly, (2) tubes and tubesheets, (3) piping and isolation values to the HPS, and (4) moisture detection and isolation valves. The NRC staff found the application of these PDCs to the selected SGS components acceptable as they comprise the portion of the HPB contained in the SGS. These pressure boundary components are NSRST, and with the exception of the tubes, main steam tube sheet, main steam nozzle, and steam pup piece, these components are low temperature as defined in ASME.III.5 and do not require any supplementation to existing code requirements or environmental testing. Supplementation of existing code requirements for the high temperature components is described in the staff evaluation of PDC 1.

The PSAR states that pressure retaining parts of the SGS support PDCs 14 and 31, PDC-CDCs 15 and 16, and associated PSFs through use of ASME.III.5, as supplemented in the materials qualification plan, as the primary means of assuring HPB integrity. In addition, ASME.III.5 rules support HPB integrity by accounting for residual stresses and requiring construction weld examination. Normal operation and AOOs map to service levels A, B, C, and D and will be provided with the OL application. Uncertainties in the material properties of the SGS SSCs will be evaluated as part of the materials qualification program, which is evaluated in SE section 3.8. Moisture ingress from water/steam is met via the application of PDC-RFDC 30, evaluated below.

The NRC staff finds that the use of ASME III.5, as assessed in SE section 3.8, will ensure that appropriate service level conditions and bounding load combinations are accounted for in the design. The materials qualification program will ensure that considerations for areas that require supplementation of ASME III.5 as well as inservice degradation of materials properties are adequately accounted for, as discussed in PDC 1. Therefore, the NRC staff finds that the preliminary design of the SGS meets the requirements of PDC 14, PDC-CDC 15, PDC-CDC 16, and PDC 31.

Integrity of Reactor Helium Pressure Boundary (PDC-RFDC 30)

The SGS meets the requirements of PDC-RFDC 30 by the design of the NISI, NIFI, and safety relief valves isolating the main steam and feedwater systems from the SGS and limiting water and/or steam ingress into the HPB. Therefore, the NRC staff finds that the preliminary design of the SGS meets the requirements of PDC-RFDC 30.

Integrity of Reactor Helium Pressure Boundary (PDC-CDC 30)

The NRC staff reviewed the application of PDC-CDC 30 to the following components 1) the SGPV assembly, (2) tubes and tubesheets, (3) piping and isolation values to the HPS, and (4) moisture detection and isolation valves. The NRC staff found the application of these PDCs to the selected SGS components acceptable as they comprise the portion of the HPB contained in the SGS.

PSAR section 6.4.12.1 states that PDC-CDC 30 is met by meeting PDC-CDC 13 as the features that support HPB leakage detection are primarily those associated with the design requirements for instrumentation and control systems to detect and isolate sources of helium leakage. The isolation valves that isolate the SGS from the HPS during depressurization events have been

categorized as risk-significant and, as described in PDC 6, will have appropriate inservice monitoring and testing to support the isolation function when necessary. Therefore, the NRC staff finds that the preliminary design of the SGS meets the requirements of PDC-CDC 30.

Fracture Prevention of Reactor Helium Pressure Boundary (PDC 31)

PDC 31 requires that the reactor helium pressure boundary design reflect consideration of service temperatures; service degradation of material properties; creep; fatigue; stress rupture; conditions of the boundary material under operating, maintenance, testing, AOOs, DBEs, and DBAs; and the uncertainties in determining (1) material properties, (2) the effects of irradiation and helium composition, including contaminants and reaction products, on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws. The NRC staff reviewed the PSAR, which states that PDC 31 requirements are met by designing and constructing the SGS SSCs in accordance with ASME III.5, as endorsed by RG 1.87, Rev. 3, and as evaluated in the structural evaluation in PSAR section 3.8. Uncertainties in the material properties of the SGS SSCs will be evaluated as part of the materials qualification program in PSAR section 3.8. The NRC staff finds that the use of ASME III.5, as assessed in SE section 3.8, will ensure that appropriate service level conditions and bounding load combinations are accounted for in the design. In addition, ASME III.5 rules support PDC 31 by accounting for residual stresses and requiring construction weld examination, which reduce the likelihood of unacceptable flaws going into service. The materials qualification program will ensure that considerations for areas that require supplementation of ASME III.5 as well as inservice degradation of material properties are adequately accounted for, as discussed in PDC 1. The NRC staff finds the SGS meets the requirements of PDC 31 based on the evaluation above and the results of the NRC staff evaluation in section 3.8 of this SE.

Residual Heat Removal (PDC-CDC 34)

The NRC staff reviewed the preliminary SGS design in accordance with PDC-CDC 34 and observed that the SGS design is supported by the restoration of primary flow following a spurious trip of the circulators, active heat removal during depressurization, restoration of active heat removal following the loss of secondary flow, and the restoration of active heat removal in certain LBEs involving water/steam ingress to ensure the specified acceptable system radionuclide release design limits are not exceeded. Safety, quality, and seismic classifications as well as the structural evaluation of SGS metallic components to maintain helium and water/steam flow path integrity support these functions and are evaluated in SE section 3.8 and PDCs 1, 2, 4, and 6 above; therefore, based on the above discussion, the NRC staff finds the preliminary design of the SGS meets the requirements of PDC-CDC 34.

Safety Functions

PSAR section 6.4.12.1 states that the SGFS's NIFI and SGSS's NISI valves support performance of RSF 1.3.1, "Isolate Water/Steam Sources." In PSAR table 5.3-1, LME identified PDC-RFDCs 13, "Instrument and Control," and PDC-RFDC 30, "Integrity of Reactor Helium Pressure Boundary," as the applicable design criteria for the SGS to support RSF 1.3.1. PSAR table 6.1-2 and section 6.4.12.1 also identify the following SRDCs for the SGS RSFs.

- SRDC for RSF 1.3.1.1: The steam generator system shall be designed to isolate the main feedwater system from the steam generator and limit water ingress into the HPB to acceptable levels in DBEs and DBAs that involve water or steam ingress

- SRDC for RSF 1.3.1.2: The steam generator system shall be designed to isolate the main steam system from the steam generator and limit steam ingress into the HPB to acceptable levels in DBEs and DBAs that involve water or steam ingress

The SGS is also designed to meet PSF 1.2.2, "Maintain core heat removal through active means," PSF 1.3.2, "Remove water/steam following isolation of the water/steam source," PSF 1.3.3, "Control water/steam ingress," PSF 2.2.1, "Isolate systems connected to HPB," and PSFs 2.7.1-2.7.3, "Control the frequency of small, medium, and large depressurization events in the steam generator vessel." These PSFs are required to be performed during specific LBEs. To ensure the successful performance of these PSFs, the following CDCs are applied:

- CDC for PSF 1.2.2.2.1—The SGS accommodates restoration of primary flow following a spurious trip of the circulators during AOOs and ensure that specified acceptable radionuclide release design limits are not exceeded
- CDC for PSF 1.2.2.2.2—The SGS accommodates active heat removal during depressurization AOOs and ensure that specified acceptable system radionuclide release design limits are not exceeded
- CDC for PDC 1.2.2.2.3—The SGS accommodates restoration of active heat removal following a loss of secondary flow during AOOs and ensure that specified acceptable radionuclide release design limits are not exceeded
- CDC for PDC 1.2.2.2.4—The SGS accommodates restoration of active heat removal in AOOs that involve water/steam ingress and ensure that specified acceptable radionuclide release design limits are not exceeded
- CDC for PSF 1.3.2.1—The SG isolation valves are capable of ceasing the flow of water into SG at such a rate as to preclude lifting of the primary relief valve during AOOs that involve water/steam ingress and ensure that specified acceptable radionuclide release design limits are not exceeded
- CDC for PSF 1.3.3.1—The SG tubes are designed to minimize the likelihood of AOOs that involve water/steam ingress that initiate anticipated operational occurrences, design basis events, or design basis accidents
- CDC for PSF 2.2.1 – The SGS is designed with the capability of being isolated from the HSS during a depressurization event initiated from the HSS
- CDC for PSF 2.7.1—The SG pressure vessel is designed to minimize the likelihood of small depressurization AOOs
- CDC for PSF 2.7.2—The SGPV is designed to minimize the likelihood of medium depressurization AOOs
- CDC for PSF 2.7.3—The SGPV is designed to minimize the likelihood of large depressurization AOOs

As evaluated above regarding SGS design basis conformance with PDCs 1 through 6, PDC-10, PDC-RFDC and CDC 13, PDC 14, PDC-CDC 15, PDC-CDC 16, PDC-RFDC and CDC 30, PDC 31, and PDC-CDC 34, the staff finds that the design of the SGS provides assurance that the integrity of the HPB will be maintained with suitable margin to satisfy RSFs and PSFs identified above.

The NRC staff finds the description of the SGS safety functions to be consistent with the guidance for the content of CP applications found in section C.6.4, "Descriptions for SR SSCs,"

of NEI 21-07, as endorsed by NRC RG 1.253, because the functions have been defined with preliminary performance requirements and are supported by plans for further development and performance validation of the final design.

Technical Specifications

Part IV of the CP application, "Technical Specifications," identifies technical specifications for the SGS based on the preliminary design and safety analysis. The proposed list of LCOs identified in the proposed technical specifications includes main steam and feedwater temperature, main steam and feedwater pressure, HSS moisture, helium to feedwater mass flow ratio, and isolation valve status as LCOs to support RSF 1.3.1.

Based on the information provided in the PSAR, the NRC staff finds the preliminary information regarding technical specifications for the SGS addresses the requirements of 10 CFR 50.36(c)(2), "Limiting Conditions for Operations." At the CP stage there are no safety limits in accordance with 10 CFR 50.36(c)(1)(ii)(A), "Limited Safety System Settings," or surveillance requirements in accordance with 10 CFR 50.36(c)(3), "Surveillance Requirements," identified for the SGS. LME states that surveillance requirements for each LCO will be provided in the OL application, as required by 10 CFR 50.36(c)(3).

Based on the review of the information provided in the CP application, as discussed above, the NRC staff finds the information provided for the SGS meets 10 CFR 50.34(a)(5) because the variables, conditions, and items that are likely to become subjects of technical specifications for the SGS are identified. Consistent with 10 CFR 50.34(b)(6)(vi), the NRC staff will evaluate the proposed technical specifications when detailed design information is available at the OL application stage.

6.4.12.2 Conclusion

The NRC staff concludes that the preliminary SGS is acceptable and meets the requirements of 10 CFR 50.34(a)(3)(i), 10 CFR 50.34(a)(3)(ii), 10 CFR 50.34(a)(3)(iii), and 10 CFR 50.34(a)(5). This conclusion is based on the following:

- The applicant identifies the appropriate PDCs for the SGS and describes the preliminary design of the SGS.
- The applicant describes how the SGS meets applicable PDCs.
- The applicant provides sufficient information to demonstrate that the preliminary design supports SR RSFs and NSRST PSFs.
- The applicant provides the materials of construction, material qualification plans, and general arrangement of the SGS.
- The applicant identified key SGS variables and conditions as probable subjects of technical specifications.

6.4.13 Reactor Protection System

PSAR section 6.4.13 describes the reactor protection system (RPS). The RPS accepts and conditions input signals from the NIS, processes trip determination and actuation voting logic, and generates output signals to initiate the required combination of the following protective actions:

- Reactor trip
- Helium circulator system (HCS) trip
- Steam generator system (SGS) isolation

The RPS is a subsystem of the plant control and data acquisition system (PCDAS), described in PSAR section 1.1.4.3.4, "Plant Control and Data Acquisition System." The RPS boundary is contained within its system-specific equipment cabinets and extends from the system input terminations that connect to the nuclear instrumentation system (NIS), described in PSAR section 6.4.14, to the system output terminations that connect to field actuation devices.

The RPS is designed to initiate protective actions when process variables monitored by the NIS reach or exceed specified setpoints to ensure safe shutdown via inherent and passive design features.

The RPS is designed to meet the fundamental instrumentation and control (I&C) design principles: independence, redundancy, diversity in support of DID, and deterministic behavior (repeatability and predictability). The RPS is built on the highly integrated protection system (HIPS) platform, which is a logic-based hardware system and has been previously reviewed and approved by the NRC staff. The HIPS topical report TR-1015-18653, "Design of the Highly Integrated Protection System Platform," Rev. 2 (ML17256A892-Public, ML17256A894-Proprietary) has been incorporated by reference in its entirety into the PSAR.

All RPS components are installed in the FHAB and are designed so that any system failure defaults to a safe state. No operator actions are needed for the RPS to support RSFs. While the RPS protective actions can be initiated manually from non-SR switches located in the MCR or locally at the RPS cabinets, the manual actuation is not credited in the DBA analysis.

System specific regulatory requirements applicable to this system are as follows:

- 10 CFR 50.34(a)(3),
- 10 CFR 50.34(a)(5),
- and
- 10 CFR 50.55a(h)(3), which incorporates by reference IEEE 603-1991

System specific guidance applicable for the evaluation of the RPS is as follows:

- "Design Review Guide (DRG): Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews" ML21011A140
- IEEE 7-4.3.2-2016, as endorsed by RG 1.152
- IEEE 336-2020, as endorsed by RG 1.30
- ANSI/ISA 67.04.01-2018, as endorsed by RG 1.105.

In addition to the applicable PDCs 1 through 6, the following system specific PDCs are applicable to the RPS:

- PDC 10 - Reactor Design
- PDC-RFDC 13 - Instrumentation and Control
- PDC 19 – Control Room
- PDC-RFDC 20 - Protection System Functions
- PDC 21 - Protection System Reliability and Testability
- PDC 22 - Protection System Independence
- PDC 23 - Protection System Failure Modes
- PDC 24 - Separation of Protection and Control Systems
- PDC 25 - Protection System Requirements for Reactivity Control Malfunctions
- PDC-RFDC 26 - Reactivity control systems
- PDC 28 - Reactivity Limits
- PDC RFDC 30 - Integrity of Reactor Helium Pressure Boundary
- PDC 31 - Fracture Prevention of Reactor Helium Pressure Boundary

6.4.13.1 Technical Evaluation

Safety Significance

The RPS is the SR control system for the Xe-100, monitoring the parameters necessary to detect LBEs that require protective actions to be performed as part of the RSFs identified through the LMP process. At the preliminary design stage, the monitored parameters include neutron flux, HPB pressure, helium-to-feedwater mass flow ratio, helium temperature, and HPB humidity. When one or more of these parameters exceed a setpoint, one or more of the three protective actions noted above would be taken.

The HCS trip is required to ensure that temperature in the reactor system remains high to allow for the inherent reactivity to shut down the reactor. If the circulators are not tripped, the temperature of the system will decrease, and the inherent reactivity feedback relied on as the primary shutdown mechanism will not be effective at achieving a low power or shutdown state. As discussed in section 6.4.2.1 of this SE, the reliance on maintaining a high temperature for this shutdown mechanism to be effective results in the fundamental safety functions of controlling reactivity opposing the fundamental safety function of controlling heat removal. Section 6.4.2.1 of this SE discusses the NRC staff's evaluation of this safety aspect.

The SGS isolation is required to limit the amount of moisture ingress into the system to prevent and mitigate radionuclide release from the fuel pebbles. Much of the fuel pebble is graphite and high humidity combined with high temperature in the system will degrade the graphite layers that are serving as part of the functional containment barrier to limit radionuclide release during certain LBEs. Failure of the SGS isolation could lead to significant increases in consequence for multiple LBEs, resulting in this being an SR function (i.e., RSF).

In section 5.4 of the SE, the NRC staff determined that the classification of the RPS as SR is consistent with the guidance in NEI 18-04, as endorsed with clarifications by RG 1.233.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR section 6.4.13.1, which states that the RPS is subject to quality assurance requirements to comply with PDC 1 and LME's Quality Assurance Program, as described in PSAR section 8.1. Additionally, staff reviewed Table 6.4.13-1, which identifies the preliminary codes and standards applicable to the RPS. These codes and standards include ASME NQA-1, IEEE 7-4.3.2, and ANSI/ISA 67.04.01, which are endorsed in RG 1.28, RG 1.152, and RG 1.105, respectively. IEEE 603 is also identified in Table 6.4.13-1 as applicable to the RPS and supports meeting 10 CFR 50.55a(h). The NRC staff evaluation of IEEE 603, as it related to the RPS, is discussed later in this section of the SE. Based on the information provided in the PSAR, the staff finds the preliminary design of the RPS meets PDC 1 by using nuclear standards endorsed in NRC regulatory guidance and the LME Quality Assurance Program.

Design Basis for Protection against Natural Phenomena (PDC 2)

The NRC staff reviewed PSAR section 6.4.13.1, which states that the RPS is designed to withstand the effects of natural phenomena, and PSAR sections 6.1.1 and 6.1.3, which describe the methods used to design the RPS to withstand the effects of natural phenomena. The RPS is protected against natural phenomena by its location within the FHAB, as described in PSAR Section 7.3.1 and evaluated in section 7.3 of this SE. The FHAB provides protection from applicable natural phenomena including earthquakes, tornadoes and associated missiles, external flooding, and extreme climate conditions and is discussed in PSAR section 3.7 and evaluated in section 3.7 of this SE. Based on the NRC staff's evaluation, including the evaluations in sections 3.7 and 7.3 of this SE, the NRC staff finds that locating the RPS in the FHAB meets PDC 2 for the preliminary design.

Fire Protection (PDC 3)

The NRC staff reviewed PSAR section 6.4.13.1, which states that the RPS is designed, located, and configured to minimize the probability and effects of fires and explosions. Configuring the RPS to minimize the probability and effects of fires and explosions includes the use of independent and redundant features. Each module's RPS is located in its own cabinet room within the FHAB. The cabinet room walls, doors, and fire barriers are rated and configured as determined by the FHA and FSSA. Further, the RPS is protected by fire protection systems tested in accordance with applicable NFPA standards and as determined by the FHA and FSSA to be necessary to protect the safety-significant SSCs and ensure performance of safe-shutdown functions. The fire protection system is one part of the plant FPP, which is designed to meet 10 CFR 50.48(a)(1). The FPP is evaluated in section 8.6 of this SE. Based on the NRC staff's review and the evaluation in section 8.6 of this SE, the staff finds the preliminary design of the RPS meets PDC 3.

Environmental and dynamic effects design bases (PDC 4)

The NRC staff reviewed PSAR section 6.4.13.1, which states that the RPS is designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and applicable LBEs. The location of the RPS in the FHAB protects the safety-significant components of the RPS from external hazards and dynamic effects and missiles resulting from equipment failure, as discussed in section 7.3 of this SE. The FHAB also provides a mild environment for the RPS, which is designed and qualified

for the expected environmental service conditions. Based on the NRC staff's review and the evaluation of the FHAB in section 7.3 of this SE, the staff finds that locating the RPS in the FHAB meets PDC 4 for the preliminary design.

Sharing of structures, systems, and components (PDC 5)

The NRC staff reviewed PSAR section 6.3.13.1, which states that each reactor module features a dedicated RPS. Based on the NRC staff's review, the NRC staff finds that the preliminary design of the RPS meets PDC 5 since the RPS is not shared between reactor modules.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR section 6.4.13.1, which states that the RPS is designed to permit appropriate monitoring and functional testing in accordance with the maintenance program, as described in PSAR section 8.3 and evaluated in section 8.3 of this SE. Additionally, the RPS design provides access for implementing preservice and inservice requirements as well as indications from online monitoring techniques to notify control room operators. The RPS is designed to permit periodic testing based on criteria provided in IEEE 336-2020, as endorsed by RG 1.30; periodic surveillance testing in accordance with IEEE 338-2022; periodic surveillance calibration and testing in accordance with IEEE 603-2018, which supports 10 CFR 50.55a(h); and IEEE 7-4.3.2-2016, as endorsed by RG 1.152. The RPS also has self-test features and is subject to technical specifications and periodic surveillances. Finally, pre-operational testing and startup testing is described in PSAR section 12.1.1 and evaluated in section 12.1.1 of the SE. Based on the NRC staff's review of the information provided in the PSAR, the evaluations in section 8.3 and 12.11 of this SE, and by using nuclear standards endorsed in NRC regulatory guidance, the NRC staff finds the preliminary design of the RPS meets PDC 6.

Reactor Design (PDC 10)

The NRC staff reviewed PSAR section 6.4.13.1, which states that the RPS is designed with appropriate margin to ensure that SARRDLs are not exceeded during any condition of normal operation, including the effects of AOOs, unmitigated by the IPS or DCS. Ensuring these limits are not exceeded is accomplished by continuously monitoring plant conditions in order to automatically initiate appropriate systems at established setpoints. As discussed in PSAR section 6.4.13.2, setpoints are established in accordance with ANSI/ISA 67.04.01-2018, as endorsed by RG 1.105, which provides a basis for establishing setpoints with appropriate margin. The RPS design also accounts for a single malfunction of the reactivity control system and the capability to command insertion of moveable poisons to maintain safe shutdown. The NRC staff notes that SARRDLs were not included in the CP application, but the NRC staff reviewed the PSAR dose consequence results for AOOs for a preliminary assessment, considering margin to the F-C target.

Based on the applicant's use of continuous monitoring of plant conditions to automatically initiate appropriate systems and the use of endorsed standards to establish the setpoints, the NRC staff finds the preliminary design of the RPS meets PDC 10.

Instrumentation and control (PDC-RFDC 13)

The NRC staff reviewed PSAR section 6.4.13.1, which states that the RPS monitors variables associated with RSFs over their anticipated ranges during normal operations and LBEs, as appropriate, to support and provide indication of RSFs. The RPS accepts and conditions input

signals from the NIS in order to monitor variables and, when variables reach or exceed specified setpoints, automatically generates output signals to initiate the appropriate combination of protective actions required to mitigate DBEs and DBAs. Therefore, the NRC staff finds the preliminary design of the RPS has instrumentation and controls necessary to accomplish the RSFs to meet PDC-RFDC 13.

Control Room (PDC 19)

The NRC staff reviewed PSAR section 6.4.13.1, which states that the RPS design includes manual initiation switches in the MCR and locally at the RPS cabinets, which provide operators with the capability to initiate prompt safe shutdown from both inside and outside the MCR. Based on the NRC staff's review, the staff finds that the preliminary design, to include specified manual switches to initiate prompt safe shutdown outside the MCR, meets the applicable portions of PDC 19 requiring equipment outside the MCR to initiate prompt safe shutdown of the reactor.

Protection system functions (PDC-RFDC 20)

The NRC staff reviewed PSAR section 6.4.13.1, which states that the RPS is designed to continuously monitor plant conditions in order to automatically initiate the operation of appropriate systems, including the reactivity control systems, to ensure that SARRDLs are not exceeded during AOOs unmitigated by the IPS or DCS, and to initiate the operation of necessary steps to support RSFs when conditions requiring initiation are sensed. Ensuring that design limits are not exceeded and RSFs are achieved is accomplished by having instrumentation and control, as described above in the evaluation of PDC-RFDC 13. Additionally, as discussed in PSAR section 6.4.13.2, operational setpoints are established in accordance with ANSI/ISA 67.04.01-2018, which is endorsed by RG 1.105. Based on the staff's review and the use of ANSI/ISA 67.04.01-2018, the staff finds the preliminary design of the RPS adequately addresses PDC-RFDC 20.

Protection System Reliability and Testability (PDC 21)

The NRC staff reviewed PSAR section 6.4.13.1, which states that the RPS is designed with sufficient redundancy and independence to ensure high functional reliability. The RPS is divided into redundant and independent divisions and arranged with voting logic to contribute to high system reliability and allow for in-service testability. Individual components of the RPS may be removed from service without loss of any RSF. The RPS is designed to permit periodic functional testing during reactor operation, including a capability to test channels independently to determine failures and losses of redundancy not identified by self-diagnostics. Also, the design of the system in accordance with IEEE 603-2018 supports compliance with PDC 21 because the standard includes related criteria requiring quality, independence, testing and calibration, repair, reliability and manual control. Based on the NRC staff's review of the reliability and testability information in addition to the redundant and independent design features, the NRC staff finds that the preliminary design of the RPS meets PDC 21.

Protection System Independence (PDC 22)

The NRC staff reviewed PSAR section 6.4.13.1, which states that the RPS is designed with four independent and redundant divisions with functional and component diversity to ensure that the

effects of natural phenomena, and of normal operating, maintenance, testing, and design basis accident conditions on redundant channels do not result in loss of a protection function. The NRC staff reviewed the preliminary RPS architecture in PSAR figure 6.4.13-1 and confirmed the architecture is configured with independent divisions. As specified in the staff evaluation of the HIPS TR, the HIPS platform supports physical, electrical, communications and functional independence, and upon successful demonstration of the ASAIs, an I&C system implemented on the HIPS will satisfy the independence requirements. Also, the design of the system in accordance with IEEE 603-2018 supports compliance with PDC 22 because the standard includes related criteria requiring quality, equipment qualification and independence. Based on the NRC staff's review of the independent and diverse features and the evaluation of the HIPS TR, the NRC staff finds the preliminary design of the RPS meets PDC 22.

Protection System Failure Modes (PDC 23)

The NRC staff reviewed PSAR section 6.4.13.1, which states that the RPS is designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of electrical power, or adverse environmental conditions are experienced. The RPS is designed such that it enables normal operation of the plant with active signals and if these signals are removed, components are designed to fail in a safe configuration. Self-diagnostics further contribute to satisfying this criterion by detecting faults that could affect performance of a protective action. Also, the design of the system in accordance with IEEE 603-2018 supports compliance with PDC 23 because the standard includes related criteria requiring single-failure tolerance, quality, system integrity, independence, reliability and common-cause failure tolerance. Based on the NRC staff's review of the information provided in the PSAR on the failure modes of the RPS, the NRC staff finds the preliminary design of the RPS meets PDC 23.

Separation of Protection and Control Systems (PDC 24)

The NRC staff reviewed PSAR section 6.4.13.1, which states that the RPS is functionally independent from the control systems. Accordingly, the RPS is designed with sufficient separation from the control systems such that all reliability, redundancy, and independence requirements are satisfied in the event of a failure of any single control system component or channel, or failure or removal from service of any single component or channel that is common to the RPS and control systems. The staff also reviewed PSAR figure 6.4.13-3, which depicts the use of isolation devices and one-way communication features between the protection system and control system. Additionally, the design of the system in accordance with IEEE 603-2018 supports compliance with PDC 24 because the standard includes related criteria requiring single failure protection and independence. Based on the NRC staff's review of the separation of protection and control systems provided in the PSAR, the NRC staff finds the preliminary design of the RPS meets PDC 24.

Protection System Requirements for Reactivity Control Malfunctions (PDC- 25)

The NRC staff reviewed PSAR section 6.4.13.1, which states that the RPS is designed to ensure that SARRDLs are not exceeded during any AOOs unmitigated by the IPS or DCS, accounting for a single malfunction of the reactivity control systems. The RPS performs this function to ensure that analytical limits established in the safety analysis are not exceeded and does this independently of the control system. As discussed in PSAR section 6.4.13.2,

operational setpoints are established in accordance with ANSI/ISA 67.04.01-2018, which is endorsed by RG 1.105 and ensures these analytical limits are not exceeded. Also, the design of the system in accordance with IEEE 603-2018 supports compliance with PDC 25 because the standard includes related criteria requiring single failure protection, independence and reliability. Based on the NRC staff's review of the information provided in the PSAR and the use of ANSI/ISA 67.04.01-2018, the NRC staff finds the preliminary design of the RPS meets PDC 25.

Reactivity control systems (PDC-RFDC 26)

The NRC staff reviewed PSAR section 6.4.13.1, which states that the RPS is designed to command insertion of moveable poisons to maintain safe shutdown when conditions requiring insertion are sensed and that the RPS performs these functions to ensure that analytical limits established in the safety analysis are not exceeded. The RCSS is described in PSAR Section 6.4.10 and consists of two independent rod banks capable of shutting down the reactor and maintaining safe shutdown. The RPS removes electrical power from the CRDM drive motors to allow the banks of the NARs to insert under the force of gravity. Based on the NRC staff's review of the capability of the RPS to command insertion of rods to maintain safe shutdown, the NRC staff finds the preliminary design of the RPS adequately addresses PDC-RFDC 26.

Reactivity Limits (PDC 28)

The NRC staff reviewed PSAR section 6.4.13.1, which states that the RPS actuation setpoints are designed to limit the potential amount and rate of reactivity increase to ensure that design basis events and design basis accidents can neither 1) result in damage to the reactor helium pressure boundary greater than limited local yielding nor 2) sufficiently disturb the core, its support structures, or other reactor vessel internals to impair significantly the capability to cool the core. The RPS performs these functions to ensure that analytical limits established in the safety analysis are not exceeded. As discussed in PSAR section 6.4.13.2, operational setpoints are established in accordance with ANSI/ISA 67.04.01-2018, which is endorsed by RG 1.105 and ensures these analytical limits are not exceeded. Based on the NRC staff's review of the information provided in the PSAR and the use of ANSI/ISA 67.04.01-2018, the NRC staff finds the preliminary design of the RPS meets PDC 28.

Integrity of reactor helium pressure boundary (PDC-RFDC 30)

The NRC staff reviewed PSAR section 6.4.13.1, which states that the RPS is designed to automatically isolate sources of moisture ingress when conditions requiring isolation are sensed. This is accomplished by having instrumentation and control, as described above in the evaluation of PDC-RFDC 13. The RPS performs these functions to ensure that analytical limits established in the safety analysis are not exceeded. As discussed in PSAR section 6.4.13.2, operational setpoints are established in accordance with ANSI/ISA 67.04.01-2018, which is endorsed by RG 1.105 and ensures these analytical limits are not exceeded. Based on the NRC staff's review of the RPS automatic isolation capabilities and the use of ANSI/ISA 67.04.01-2018, the NRC staff finds the preliminary design of the RPS meets PDC-RFDC 30.

Summary of Compliance with Applicable PDCs

The NRC review determined that the preliminary information provided to support compliance with applicable PDCs for the RPS is adequate at this stage of the design and sufficient to support the CP application.

Safety Functions

PSAR section 6.4.13.1 states that the RPS is designed to support RSFs 1.1.1 (Control reactivity with inherent reactivity feedback), 1.1.2 (Maintain long-term subcriticality), and 1.3.1 (Isolate water/steam sources). LME identified PDC-RFDC 13 (Instrumentation and control), PDC-RFDC 20 (Protection system functions), PDC-RFDC 26 (Reactivity control systems), and PDC-RFDC 30 (Integrity of the reactor helium pressure boundary) as applicable design criteria for the RPS to support RSFs 1.1.1, 1.1.2, and 1.3.1. In addition, LME identified PDCs 1 through 6 as cross-cutting and/or programmatic design requirements with which the RPS design complies. The NRC staff finds that these PDC selections align with the RSFs identified for the preliminary design of RPS because they are consistent with PSAR table 5.3-1, which maps PDCs with RSFs based on PRA results.

Based on the PDC and safety function evaluations, described above, the NRC staff finds that the preliminary design of the RPS provides adequate support for RSFs 1.1.1, 1.1.2, and 1.3.1.

Evaluation of Additional Requirements and Criteria

Technical Specifications

The RPS has been identified to have technical specifications and surveillance requirements based on the preliminary design and the associated preliminary safety analysis, with additional details provided in Part IV of the CP application, “Technical Specifications.” The proposed list of preliminary systems subject to LCOs are identified in PSAR Part IV, Table IV-3. The RPS trip variables for reactor trips, HCS trips, and SGS isolation are listed as initial LCOs to support RSFs 1.1.1, 1.1.2, and 1.3.1. Additionally, the RPS uses a set of trip parameter setpoints and logic to initiate necessary protective actions prior to exceeding any analytical limits (ALs) to protect safety limits (SLs). The NRC staff reviewed PSAR table 6.4.13-2, which proposes that the initial set of trip parameters and setpoints be calculated in accordance with ANSI/ISA 67.04.01-2018, as endorsed by RG 1.105. The RPS is also designed to meet surveillance requirements to ensure that the necessary quality is maintained such that operation is within safety limits and LCOs are met. As stated in Part IV of the CP application, information on specific LSSS values and surveillance requirements are being developed using the methodology in NEI 18-04 and guidance in DANU-ISG-2022-08 and will be provided in the TS to support the OL application.

Based on its review of the information provided in the PSAR, the NRC staff finds the information provided for the RPS meets 10 CFR 50.34(a)(5) because the variables, conditions, and items that are likely to become subjects of technical specifications for the RPS are identified. Consistent with 10 CFR 50.34(b)(6)(vi), the NRC staff will evaluate the proposed technical specifications when detailed design information is available at the OL application stage.

Limitations and Conditions on use of the HIPS platform

The RPS is based on the NRC-approved HIPS platform. The NRC staff review is documented in the approved TR referenced earlier in this subsection. As discussed in the approved TR, there are application-specific action items (ASAIs), which applicants and licensees that reference the TR are required to address. PSAR table 1.4-15 states that LME has committed to addressing the ASAIs and to supplement the HIPS TR as part of the OL application.

Based on the NRC staff's review of the information provided in the PSAR, the NRC staff finds the approach to address ASAIs as part of a future OL application to be reasonable because ASAIs are not required to be addressed at the CP stage. The staff would verify that the ASAIs have been addressed as part of its review of an OL application.

Codes and Standards - Safety Systems

10 CFR 50.55a(h)(3) identifies, in part, that construction permits must meet the requirements for safety systems in IEEE Std 603-1991 and the correction sheet dated January 3, 1995. The HIPS TR includes ASAI #3, which states that an applicant or licensee referencing the TR must identify the approach taken to satisfy each applicable clause of IEEE Std 603-1991. The NRC staff notes that PSAR table 6.4.13-1 provides industry codes and standards that are preliminarily identified as applicable to the RPS and that the table includes IEEE Std 603-2018. Additionally, the PSAR identifies deviations and exceptions to IEEE 603-2018 and IEEE 7-4.3.2-2016. The PSAR states that LME's preliminary review of the 1991 and 2018 versions of IEEE 603 concludes that compliance with 10 CFR 50.55a(h)(3) can be demonstrated, as the 2018 version meets or exceeds the requirements of IEEE 603-1991. LME states a comparison, including evaluation of and provision of bases for acceptability of the differences between the two versions will be provided to support an OL application. The staff would evaluate the deviations and exceptions as part of its review of an OL application.

The current NRC position on common-cause failure (CCF) is provided in SRM-SECY-22-0076. The NRC staff reviewed PSAR section 6.4.13.2, which states that the I&C diversity and defense-in-depth assessment is encompassed by the overall plant defense-in-depth analysis. The PSAR also states that CCF is addressed through the implementation of the risk-informed, performance-based approach described in NEI 18-04 and by designing in accordance with consensus industry codes and standards.

Based on the NRC staff's review of information provided in the PSAR, consistent with 50.34(a)(3)(iii), the staff finds the information provided on how the RPS can meet 10 CFR 50.55a(h)(3) and the approach to addressing CCF is sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety. As part of a more detailed evaluation of the RPS during the staff's review of a future OL application, the staff will confirm that the final design meets the requirements of 10 CFR 50.55a(h)(3), evaluate acceptability of any proposed deviations and exceptions, evaluate how SRM-SECY-22-0076 is addressed, and confirm that ASAI #3 is addressed.

Fundamental I&C principles

The DRG provides guidance for a reliable and robust I&C system. To achieve a reliable and robust I&C system, the I&C system design should meet the fundamental I&C design principles

of independence, redundancy, diversity in support of DID, and deterministic behavior (repeatability and predictability).

In PSAR section 6.4.13.1, LME states that the RPS is designed to meet these fundamental I&C design principles. As documented in the NRC staff evaluation and approval of the HIPS platform in the HIPS TR, the NRC staff determined that the HIPS platform is designed to meet these fundamental design principles and the NRC staff confirmed these fundamental principles are reflected in the preliminary RPS architecture shown in PSAR figure 6.4.13-1. Additionally, the PDCs applicable to the RPS and criteria from IEEE 603-2018 further support these fundamental design principles.

Based on the NRC staff's review of the commitment to meet these fundamental I&C principles, the use of the approved HIPS platform, and designing the RPS in conformance with IEEE 603-2018, the NRC staff finds that the preliminary design of the RPS meets 10 CFR 50.34(a)(3)(iii) by providing a preliminary architecture sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.

6.4.13.2 Conclusion

Based on the review described above, the NRC staff concludes that the information provided in PSAR section 6.4.13 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter. Specifically, the information provided:

- meets the requirements of 10 CFR 50.34(a)(3) for the RPS by describing the preliminary design identifying the applicable PDCs, by describing the design bases and the relationship between the design bases and the PDCs, and by describing information relative to the general architecture and configuration sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety;
- meets the requirements of 10 CFR 50.34(a)(5) for the RPS by identifying key design features as probable subjects of technical specifications; and
- is sufficient to show how the RPS can meet 10 CFR 50.55a(h)(3).

Additionally, the PSAR commits to addressing HIPS TR ASAIs and will be reviewed by the staff during the OL application. The NRC staff also finds the design of the RPS is consistent with guidance in the DRG, including adherence to the fundamental I&C design principles. Therefore, the NRC staff concludes any information not available at this stage can be reasonably left to be evaluated at the OL application stage in accordance with 10 CFR 50.35(a) and that the preliminary information provided for the RPS is adequate to support issuance of a CP. A detailed evaluation of the RPS will be performed during OL application stage, at which time the NRC staff will confirm that the final design meets applicable regulations.

6.4.14 Nuclear Instrumentation System

PSAR section 6.4.14 describes the nuclear instrumentation system (NIS). The NIS converts measured process variables into corresponding signals, which are supplied to the RPS, IPS, DCS, and PEMS.

The NIS is a subsystem of the PCDAS, described in PSAR section 1.1.4.3.4, "Plant Control and Data Acquisition System." The NIS boundary extends from the sensing device or element interconnections to the interconnections that supply signals to the RPS as described in PSAR section 6.4.13, to the IPS as described in PSAR section 7.3.11, to the DCS described in PSAR section 7.3.12, and to the PEMS described in PSAR Section 1.1.4.

Key design features of the NIS meet the identified system-specific SRDCs and CDCs and support performance of the identified RSFs and NSRST PSFs. The NIS is designed to sense conditions requiring protective actions when monitored process variables reach or exceed specified setpoints to ensure safe shutdown via inherent and passive design features. The NIS supplies signals to the RPS to support performance of the RSFs and supplies signals, via isolated, unidirectional outputs to the IPS, DCS, and PEMS to support performance of both NSRST PSFs and NST PSFs.

The NIS is designed to meet the fundamental instrumentation and control (I&C) design principles: independence, redundancy, and diversity in support of DID, and deterministic behavior (repeatability and predictability). All NIS components are installed in the SST.

System specific regulatory requirements are as follows:

- 10 CFR 50.34(a)(5),
- and
- 10 CFR 50.55a(h)(3), which incorporates by reference IEEE 603-1991

System specific guidance applicable for the evaluation of the NIS is as follows:

- "Design Review Guide (DRG): Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews" ML21011A140
- IEEE 7-4.3.2-2016, as endorsed by RG 1.152
- IEEE 336-2020, as endorsed by RG 1.30
- ANSI/ISA 67.04.01-2018, as endorsed by RG 1.105.

In addition to PDCs 1 through 6, and PDC 10, the following PDCs are applicable to the NIS:

- PDC-RFDC 13 and PDC-CDC 13 - Instrumentation and Control
- PDC 20 and PDC-RFDC 20 - Protection System Functions
- PDC 21 - Protection System Reliability and Testability
- PDC 22 - Protection System Independence
- PDC 23 - Protection System Failure Modes
- PDC 24 - Separation of Protection and Control Systems
- PDC 25 - Protection System Requirements for Reactivity Control Malfunctions
- PDC-RFDC 26 and PDC-CDC 26 - Reactivity control systems
- PDC 28 – Reactivity Limits
- PDC-RFDC 30 and PDC-CDC 30– Integrity of Reactor Helium Pressure Boundary

- PDC-CDC 34 - Residual Heat Removal

6.4.14.1 Technical Evaluation

Safety Significance

The NIS converts measured process variables into corresponding signals, which are supplied to the RPS, IPS, DCS, and PEMS. The NIS senses key plant process variables, converts the measured process variables into corresponding signals, and supplies the signals to the RPS to support performance of RSFs. When specific conditions are detected by NIS sensors, the RPS will initiate corresponding protective actions as described in PSAR section 6.4.13. The NIS supplies signals to the IPS, DCS, and PEMS to support performance of NSRST PSFs and NST PSFs. These signals are supplied via isolated, unidirectional outputs from the NIS.

In section 5.4 of the SE, the NRC staff determined that the classification of the NIS as SR is consistent with the guidance in NEI 18-04, as endorsed with clarifications by RG 1.233. The NRC staff notes that the NIS also performs NSRST functions. Based on the information provided in the PSAR, the NRC staff finds the NIS classification as SR is reasonable.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR table 6.4.14.1, which identifies codes and standards applicable to the NIS. These codes and standards include ASME NQA-1, which is endorsed by RG 1.28, and IEEE 603, which supports meeting 10 CFR 50.55a(h). The NRC staff evaluation of IEEE 603, as it relates to the NIS, is discussed later in this section of the SE. Additionally, as stated in PSAR section 6.4.13.1, the SR NIS is subject to quality assurance requirements consistent with LME's quality assurance program, as described in PSAR section 8.1. The NRC staff evaluated LME's quality assurance program in SE section 8.1. Based on NRC staff's evaluation of the information provided in the PSAR, the staff finds the preliminary design of the NIS is consistent with PDC 1 by using nuclear standards endorsed in NRC regulatory guidance and the LME quality assurance program.

Design Basis for Protection against Natural Phenomena (PDC 2)

The NRC staff reviewed PSAR section 6.4.14.1, which states that the NIS is designed to withstand the effects of natural phenomena and that the methods to withstand such effects are described in PSAR sections 6.1.1. and 6.1.3. The NIS is protected against natural phenomena by its location within the SST, as described in PSAR section 6.4.1 and evaluated in section 6.4 of this SE. The SST provides protection from applicable natural phenomena including earthquakes, tornadoes and associated missiles, external flooding, and extreme climate conditions and is discussed in PSAR section 3.7 and evaluated in section 3.7 of this SE. Based on the NRC staff's evaluation, including the evaluations in sections 3.7 and 6.4 of this SE, the NRC staff finds that locating the NIS in the SST is consistent with PDC 2 for the preliminary design.

Fire Protection (PDC 3)

The NRC staff reviewed PSAR section 6.4.14.1, which states that the NIS is designed, located, and configured to minimize the probability and effect of fires and explosions. Each reactor module's dedicated NIS is comprised of independent and redundant divisional instrument channels that perform monitoring functions for its associated module. All four NISs are located within the SST and are physically separated from the other modules by fire-rated barriers. Within each module's NIS, equipment and cabling associated with each divisional instrument channel are separated from other divisions by fire-rated barriers. Fire barriers are rated and configured by a fire hazard analysis and post-fire shutdown analysis to minimize the likelihood of a fire simultaneously affecting multiple NIS divisions. Further, the NIS is protected by fire protection systems tested in accordance with applicable NFPA standards and as determined by the FHA and FSSA, which are part of the plant FPP that is designed to meet 10 CFR 50.48(a)(1). The fire protection systems and FPP are evaluated in section 8.6 of this SE. Based on the NRC staff's review and the evaluation in section 8.6 of this SE, the NRC staff finds the preliminary design of the NIS is consistent with PDC 3.

Environmental and dynamic effects design bases (PDC 4)

The NRC staff reviewed PSAR section 6.4.14.1, which states that the NIS is designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and applicable LBEs. The SST protects the safety-significant components of the NIS from external hazards and dynamic effects and missiles resulting from equipment failure as discussed in section 6.4.1.1.2 of this SE. The SR NIS components are qualified for the expected environmental service conditions in the SST. Additionally, protection strategies, to be developed in the FSAR, will consider the potential for adverse impact of NSRST NIS component on SR SSCs. Based on the NRC staff's review and the evaluation of the SST in section 6.4.1.1.2 of this SE, the NRC staff finds that locating the NIS in the SST is consistent with PDC 4.

Sharing of structures, systems, and components (PDC 5)

The NRC staff reviewed PSAR section 6.3.14.1, which states that each reactor module features a dedicated NIS. Based on the NRC staff's review, the NRC staff finds that the preliminary design of the NIS is consistent with PDC 5 since the NIS is not shared between reactor modules.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR section 6.4.14.1, which states that the safety-significant NIS SSCs are designed to permit appropriate monitoring and functional testing in accordance with the maintenance program, as described in PSAR section 8.3. The NIS is designed to permit periodic testing based on criteria provided in IEEE 336-2020, as endorsed by RG 1.30; periodic surveillance testing in accordance with IEEE 338-2022; and periodic surveillance calibration and testing in accordance with IEEE 603-2018, which supports 10 CFR 50.55a(h), and IEEE 7-4.3.2-2016, as endorsed by RG 1.152. The NIS SSCs subject to technical specifications undergo periodic surveillances. The NIS is also subject to pre-operational testing and startup testing as described in PSAR section 12.1.1 and evaluated in section 12.1.1 of the SE. Finally, baseline testing along with monitoring, testing, and surveillance requirements provide confidence that the capability of the NIS is retained throughout the plant's life. Based on the NRC staff's review of the information provided in the PSAR, the evaluations in sections 8.3

and 12.1.1 of this SE, and by using nuclear standards endorsed in NRC regulatory guidance, the staff finds the preliminary design of the NIS is consistent with PDC 6.

Reactor Design (PDC 10)

The NRC staff reviewed PSAR section 6.4.14.1, which states that the NIS is designed with appropriate margin to ensure that SARRDLs are not exceeded during any condition of normal operation, including the effects of AOOs. PDC 10 is a higher-level requirement that is met for the NIS design via application of PDC-RFDC 20, PDC 25, PDC-RFDC-26 and PDC-CDC 20, PDC-CDC 25 and PDC-CDC 26. The NRC staff evaluation of the specific PDCs that support meeting of PDC 10 is presented later in this section of the SE.

Based on the evaluation of the preliminary NIS design's conformance to PDC-RFDC 20, PDC 25, PDC-RFDC-26, PDC-CDC 20, PDC-CDC 25 and PDC-CDC 26, as discussed later in this section of the SE, the NRC staff finds the preliminary design of the NIS is consistent with PDC 10 applied as a high-level requirement.

Instrumentation and control (PDC-RFDC 13 and PDC-CDC 13)

The NRC staff reviewed PSAR section 6.4.14.1, which states that the NIS is designed to monitor variables over their anticipated ranges during normal operations and LBEs, as appropriate, to support and provide indication of functions performed by safety-significant SSCs. The NIS instruments supply signals to the RPS, IPS, DCS, and PEMS. Based on the NRC staff's review, the staff finds the preliminary design of the NIS to convert process variable information to the RPS, IPS, and DCS, necessary to accomplish RSFs and PSFs, is consistent with PDC-RFDC 13 and PDC-CDC 13.

Protection system functions (PDC-RFDC 20 and PDC-CDC 20)

The NRC staff reviewed PSAR section 6.4.14.1, which states that the NIS is designed to sense conditions requiring initiation of the necessary systems and components to perform RSFs and NSRST PSFs. The PSAR states that design of the NIS is in accordance with IEEE 603-2018, which supports meeting 10 CFR 50.55a(h). The NRC staff reviewed the information presented for the NIS in the PSAR and confirmed that the NIS is designed to perform functions related to sensing conditions requiring initiation of the necessary systems and components to perform required RSFs and NSRST PSFs to ensure that analytical limits established by safety analyses are not exceeded. Based on the NRC staff's review of the information provided in the PSAR, the NRC staff finds the preliminary design of the NIS to convert process variable information to the RPS and IPS to accomplish necessary RSFs and PSFs is consistent with PDC-RFDC 20 and PDC-CDC 20.

Protection System Reliability and Testability (PDC 21)

The NRC staff reviewed PSAR section 6.4.14.1, which states that the NIS is designed with sufficient redundancy and independence to ensure high functional reliability. Individual components of the NIS may be removed from service without loss of required minimum redundancy unless the acceptable reliability of operation of the system can be otherwise demonstrated. The NIS is designed to permit periodic functional testing during reactor operation, including a capability to test channels independently to determine failures and losses

of redundancy, including while in service. Also, the design of the system in accordance with IEEE 603-2018 supports compliance with PDC 21 because the standard includes related criteria requiring quality, independence, testing and calibration, repair, reliability and manual control. Based on the NRC staff's review of the reliability and testability information, in addition to the redundant and independent design features, the NRC staff finds the preliminary design of the NIS is consistent with PDC 21.

Protection System Independence (PDC 22)

The NRC staff reviewed PSAR section 6.4.14.1, which states that the NIS is designed with four independent and redundant divisions, with functional diversity to ensure that the effects of natural phenomena, and of normal operating, maintenance, testing, and design basis accident conditions on redundant channels do not result in loss of a protection function. Additionally, the NRC staff reviewed the NIS sensors identified in the preliminary RPS architecture in PSAR figure 6.4.13-1, and confirmed the architecture is configured with four independent and redundant divisions. Also, the design of the system in accordance with IEEE 603-2018 supports compliance with PDC 22 because the standard includes related criteria requiring quality, equipment qualification and independence. Based on the information provided in the PSAR, the staff finds the preliminary design of the NIS is consistent with PDC 22.

Protection System Failure Modes (PDC 23)

The NRC staff reviewed PSAR Section 6.4.14.1, which states that the NIS is designed to fail in a safe state or in a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of electrical power, or adverse environmental conditions are experienced. The NIS is designed such that it supplies signals to the RPS, which enables normal operation of the plant with active signals and if these signals are removed, components are designed to fail in a safe configuration. Also, the design of the system in accordance with IEEE 603-2018 supports compliance with PDC 23 because the standard includes related criteria requiring single-failure tolerance, quality, system integrity, independence, reliability and common-cause failure tolerance. Based on the NRC staff's review of the information provided in the PSAR on the failure modes of the NIS, the NRC staff finds the preliminary design of the NIS is consistent with PDC 23.

Separation of Protection and Control Systems (PDC 24)

The NRC staff reviewed PSAR section 6.4.14.1, which states that the NIS is independent and has sufficient separation from the control systems such that all reliability, redundancy and independence requirements associated with PDC 24 are satisfied in the event of a failure of any single control system, component, or channel or removal from service of any single component or channel that is common to NIS and control systems. All signals supplied from the NIS to the control systems are via isolated, unidirectional pathways. Also, the design of the system in accordance with IEEE 603-2018 supports compliance with PDC 24 because the standard includes related criteria requiring single failure protection and independence. Based on the NRC staff's review of the separation of protection and control systems provided in the PSAR, the NRC staff finds the preliminary design of the NIS is consistent with PDC 24.

Protection System Requirements for Reactivity Control Malfunctions (PDC 25)

The NRC staff reviewed PSAR section 6.4.14.1, which states that the NIS is designed to ensure that SARRDLs are not exceeded during any AOOs, accounting for a single malfunction of the reactivity control systems. The RPS performs this function to ensure that analytical limits established in the safety analysis are not exceeded. As discussed in PSAR section 6.4.13.2, RPS setpoints are established in accordance with ANSI/ISA 67.04.01-2018, which is endorsed by RG 1.105 and ensures these analytical limits are not exceeded. The design of the NIS system in accordance with IEEE 603-2018 supports compliance with PDC 25 because the standard includes related criteria requiring single failure protection, independence and reliability. Based on the NRC staff's review of the information provided in the PSAR, and the use of IEEE 603-2018 and ANSI/ISA 67.04.01-2018, the NRC staff finds the preliminary design of the NIS is consistent with PDC 25.

Reactivity control systems (PDC-RFDC 26 and PDC-CDC 26)

The NRC staff reviewed PSAR section 6.4.14.1, which states that the NIS is designed to sense conditions requiring insertion of moveable poisons by detecting neutron flux and supplying corresponding signals to the RPS and IPS, which initiate independent and diverse means of negative reactivity insertion. Based on the NRC staff's review of the means to insert negative reactivity with the RPS and IPS based on signals from the NIS to accomplish RSFs and PSFs, the NRC staff finds the preliminary design of the NIS is consistent with PDC-RFDC 26 and PDC-CDC 26.

Reactivity Limits (PDC 28)

The NRC staff reviewed PSAR section 6.4.14.1, which states that the NIS supplies process variable signals to the RPS, which utilizes setpoints designed to limit the potential amount and rate of reactivity increase to ensure that DBEs and DBAs can neither (1) result in damage to the reactor helium pressure boundary greater than limited local yielding, nor (2) sufficiently disturb the core, its support structures, or other reactor vessel internals to impair significantly the capability to cool the core. The NIS is designed to perform this function to ensure that analytical limits established in the safety analysis are not exceeded. As discussed in PSAR section 6.4.13.2, RPS setpoints are established in accordance with ANSI/ISA 67.04.01-2018, which is endorsed by RG 1.105 and ensures these analytical limits are not exceeded. Based on the NRC staff's review of the information provided in the PSAR and the use of ANSI/ISA 67.04.01-2018, the NRC staff finds the preliminary design of the NIS is consistent with PDC 28.

Integrity of reactor helium pressure boundary (PDC-RFDC 30 and PDC-CDC 30)

The NRC staff reviewed PSAR section 6.4.14.1, which states that the NIS is designed to sense moisture ingress into the HPB by detecting neutron flux and humidity and supplying corresponding signals to the RPS, which initiates SG isolation. The NIS is designed to sense abnormal helium egress from the HPB by detecting pressure and supplying corresponding signals to the IPS, which initiates HPB isolation. Based on the NRC staff's review of the RPS and IPS isolation capabilities supported by NIS signals to accomplish RSFs and PSFs, the NRC staff finds the preliminary design of the NIS is consistent with PCD-RFDC 30 and PDC-CDC 30.

Residual Heat Removal (PDC-CDC 34)

The NRC staff reviewed PSAR section 6.4.14.1, which states the NIS is designed to support safety-significant SSCs that provide an active means to transfer fission product decay heat and other residual heat from the reactor core to the ultimate heat sink at a rate such that specified acceptable system radionuclide design limits are not exceeded during AOOs by detecting helium temperature and mass flow and supplying corresponding signals to the DCS, which initiates the appropriate heat removal control functions. Based on the NRC staff's review of the residual heat removal features supported by NIS signals to accomplish PSFs, the NRC staff finds the preliminary design of the NIS is consistent with PDC-CDC 34.

Summary of Compliance with Applicable PDCs

The NRC review determined that the preliminary information provided can support compliance with applicable PDCs for the NIS to the level necessary for this stage of the design and is sufficient to support the CP application.

Safety Functions

PSAR section 6.4.14.1 states that the NIS is designed to support RSFs 1.1.1 (Control reactivity with inherent reactivity feedback), 1.1.2 (Maintain long-term subcriticality), and 1.3.1 (Isolate water/steam sources). LME identified PDC-RFDC 13, PDC-RFDC 20, PDC-RFDC 26, and PDC-RFDC 30, as applicable design criteria for the NIS to support RSFs 1.1.1, 1.1.2, and 1.3.1. The NIS is also designed to support NSRST PSFs 1.1.3 (Control reactivity with moveable poisons), 1.2.2 (Maintain core heat removal through active means), 1.3.2 (Remove water/steam following isolation of water/steam sources), and 2.2.1 (Isolate systems connected to helium pressure boundary). LME identified PDC-CDC 13, PDC-CDC 26, PDC-CDC 30, and PDC-CDC 34 as applicable design criteria for the NIS to support NSRST RSFs 1.1.3, 1.2.2, 1.3.2, and 2.2.1. The NRC staff finds that these PDC selections align with the RSFs identified for the preliminary design of NIS because they are consistent with PSAR table 5.3-1, which maps PDCs with RSFs based on PRA results. Based on the PDC and safety function evaluations, described above, the NRC staff finds that the preliminary design of the NIS provides adequate support for RSFs 1.1.1, 1.1.2, and 1.3.1 and the NSRST PSFs 1.1.3, 1.2.2, 1.3.2, and 2.2.1.

Evaluation of Additional Requirements

Technical Specifications

The NIS has been identified to have technical specifications and surveillance requirements based on the preliminary design and the associated preliminary safety analysis, with additional details provided in Part IV of the CP application, "Technical Specifications." The proposed list of preliminary systems subject to LCOs are identified in PSAR Part IV, Table IV-3. The RPS trip variables, which are supplied by the NIS, for reactor trips, HCS trips, and SGS isolation are listed as initial LCOs to support RSFs 1.1.1, 1.1.2, and 1.3.1.

Additionally, the RPS uses a set of trip parameter setpoints and logic, which are supplied by signals from the NIS, to initiate necessary protective actions prior to exceeding any analytical limits (ALs) to protect safety limits (SLs). The NRC staff reviewed PSAR table 6.4.13-2, which proposes the initial set of trip parameters. The NIS is also designed to meet surveillance requirements to ensure that the necessary quality is maintained such that operation is within safety limits and LCOs are met. As stated in Part IV of the CP application, information on

specific LSSS values and surveillance requirements are being developed using the methodology in NEI 18-04 and guidance in DANU-ISG-2022-08 and will be provided in the TS to support the OL application.

Based on the NRC staff's review of the information provided in the PSAR, the NRC staff finds the information provided for the NIS meets 10 CFR 50.34(a)(5) because the variables, conditions, and items that are likely to become subjects of technical specifications for the NIS are identified. Consistent with 10 CFR 50.34(b)(6)(vi), the NRC staff will evaluate the proposed technical specifications when detailed design information is available at the OL application stage.

Codes and Standards - Safety Systems

10 CFR 50.55a(h)(3) identifies, in part, that construction permits must meet the requirements for safety systems in IEEE Std 603-1991 and the correction sheet dated January 3, 1995. The HIPS TR includes ASAI #3, which states that an applicant or licensee referencing the TR must identify the approach taken to satisfy each applicable clause of IEEE Std 603-1991. The NRC staff notes that PSAR table 6.4.14-1 provides industry codes and standards that are preliminarily identified as applicable to the NIS and that the table includes IEEE Std 603-2018. Additionally, the PSAR identifies deviations and exceptions to IEEE 603-2018 and IEEE 7-4.3.2-2016. The PSAR states that LME's preliminary review of the 1991 and 2018 versions of IEEE 603 concludes that compliance with 10 CFR 50.55a(h)(3) can be demonstrated, as the 2018 version meets or exceeds the requirements of IEEE 603-1991. LME states a comparison, including evaluation of and provision of bases for acceptability of the differences between the two versions will be provided to support an OL application. The staff would evaluate the deviations and exceptions as part of its review of an OL application.

Based on the NRC staff's review of the information provided in the PSAR, consistent with 50.34(a)(3)(iii), the staff finds the information provided on how the NIS meets 50.55a(h)(3) is sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety. As part of a more detailed evaluation of the NIS during the staff's review of a future OL application, the staff will confirm that the final design meets the requirements of 10 CFR 50.55a(h)(3) and evaluate any proposed deviations and exceptions.

Fundamental I&C principles

The DRG provides guidance for a reliable and robust I&C system. To achieve a reliable and robust I&C system, the I&C system design should meet the fundamental I&C design principles of independence, redundancy, diversity in support of DID, and deterministic behavior (repeatability and predictability).

In PSAR section 6.4.14.1, LME states that the NIS is designed to meet these fundamental I&C design principles. The PDCs applicable to the NIS and criteria from IEEE 603-2018 further support these design principles.

Based on the NRC staff's review of the commitment to meet these fundamental I&C principles and designing the NIS in conformance with IEEE 603-2018, the NRC staff finds that the preliminary design of the NIS meets the fundamental I&C principles as discussed in the DRG.

6.4.14.2 Conclusion

The NRC staff reviewed the NIS described in PSAR section 6.4.14 and finds the information on the NIS acceptable because the PSAR adequately provides a discussion to support conformance of the NIS design with applicable PDCs, a description of how the NIS supports the PSFs, preliminary information related to technical specifications and surveillance requirements consistent with 10 CFR 50.34(a)(5), and commitments to meeting the requirements of 10 CFR 50.55a(h)(3). Additionally, the staff finds the preliminary design of the NIS is consistent with guidance in the DRG, including adherence to the fundamental I&C design principles. Therefore, the NRC staff concludes that the preliminary information provided for the NIS is adequate at this stage of the design. A detailed evaluation of the NIS will be performed during OL application stage, at which time the staff will confirm that the final design conforms to this design and meets applicable regulations.

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