U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Final Safety Evaluation

Related to the
Long Mott Energy, LLC,
Construction Permit Application for
the Long Mott Generating Station

Completed: October 2026

Office of Advanced Reactors (OAR)

ABSTRACT

This safety evaluation documents the U.S Nuclear Regulatory Commission (NRC) staff's technical review of the construction permit application submitted by Long Mott Energy, LLC (LME), under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," for the Long Mott Generating Station (LMGS). The LMGS is proposed to be built in Calhoun County, Texas.

The LMGS consists of four X Energy, LLC (X-energy) Xe-100 high-temperature gas-cooled reactors. These four Xe-100 reactors are capable of generating a total output of 800 megawatts-thermal (MWt) (or 320 megawatts-electric (MWe)). The reactors will be used to generate electricity and process steam for the Dow Chemical Company's Seadrift Operations with any excess power exported to the electrical grid.

The NRC's Advisory Committee on Reactor Safeguards independently reviewed the unique, novel, and noteworthy topics affecting the safety case and provided the results of its review to the Commission in a report dated October 16, 2026 (ML26289A141).

This safety evaluation presents the NRC staff's review of the LMGS construction permit application based on information submitted by LME through September 2026. On the basis of its review of the construction permit application, the NRC staff determined that, among other things: (1) applicable standards and requirements of the Atomic Energy Act of 1954, as amended, and the Commission's regulations have been met; (2) the application generally conforms to the applicable guidance for a construction permit applicant for a commercial non-light water reactor, with acceptable variances as noted in this safety evaluation; (3) based on the preliminary design of the facility, there is reasonable assurance that the final design will conform to the design basis with adequate margin for safety; (4) there is reasonable assurance that the facility can be constructed in conformity with the permit, the provisions of the Atomic Energy Act of 1954, as amended, and the Commission's regulations; and (5) the financial information, technical analyses, programs, and organization described in the application, as supplemented, demonstrated that LME is financially and technically qualified to engage in the construction of its proposed facility in accordance with the Commission's regulations. Therefore, the NRC staff recommends that the Commission make the necessary findings with respect to the safety of the construction permit in accordance with 10 CFR 50.35, "Issuance of construction permits;" 10 CFR 50.40, "Common standards;" and 10 CFR 50.50, "Issuance of licenses and construction permits."

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ABBREVIATIONS AND ACRONYMS

ACI	American Concrete Institute
ADAMS	Agencywide Documents Access and Management System
AEA	Atomic Energy Act
AGR	Advanced Gas Reactor
AHU	Air handling unit
AIA	Aircraft impact assessment
AISC	American Institute of Steel Construction
ALARA	As low as is reasonably achievable
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
AoA	Area of applicability
AOO	Anticipated operational occurrences
ARDP	Advanced Reactor Demonstration Program
ASAI	Application-specific action item
ASB	Access and security building
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing and Materials
AVR	Arbeitsgemeinschaft Versuchsreaktor
BDBE	Beyond design basis event
BDT	Blow down tank
BUMS	Burnup measurement system
C	Celsius
CAFTA	Computer Aided Fault Tree Analysis System
CAM	Continuous air monitor
CB	Core barrel
CCF	Common cause failure
CDC	Complementary design criteria
CDCI	CDFS chemical injection subsystem
CDFS	Conventional island condensate and feedwater system
CDFW	Feedwater system
CDS	Condenser system
CEB	Controls and electrical building
CEHV	Inter-unit access tunnel HVAC system
CFR	*Code of Federal Regulations*
CGD	Commercial grade dedication
CI	Conventional island
CICW	Conventional island cooling water system
CIE	Conventional island electrical system
CIFP	Conventional island fire protection
CIIA	Conventional island compressed and instrument air system
CMDA	Core Design Methods and Analysis
CMS	Core metallic structures
CO	Carbon monoxide
CORRM	Core corrosion calculations model
CP	Construction permit
CPA	Construction permit application

CPF	Canister processing facility
CPHV	Canister processing facility HVAC system
CRDC	Control rod drive controller
CRDM	Control rod drive mechanism
CRW	Control Rod Withdrawal
CV	Cross vessel
CVAP	Comprehensive vibration assessment program
DA	Data analysis
DANU	Division of Advanced Reactors and Non-Power Production and Utilization Facilities
DBA	Design basis accidents
DBE	Design basis events
DBHL	Design basis hazard levels
DCPV	Defuel chute pressure vessel
DCS	Distributed control system
DID	Defense-in-depth
DIV	Double isolation valve
DLOFC	Depressurized loss of forced cooling
DMA	Degradation mechanism assessment
DOE	U.S. Department of Energy
D-RAP	Design reliability assurance program
DRG	Design Review Guide
DRS	Design response spectrum
DUSTM	Dust production model
EAB	Exclusion area boundary
EFPY	Effective full power years
EM	Evaluation model
EMBC	Electromagnetic bearing controller
EMDAP	Evaluation Model Development and Assessment Process
EPRI	Electric Power Research Institute
ER	Environmental Report
ES	Event sequence
ESQ	Event sequence quantification
ET	Event tree
F	Fahrenheit
F-C	Frequency-consequence
FCP	Financial capacity plan
FFTP	Fresh fuel transport packages
FHA	Fire hazards analysis
FHAB	Fuel handling annex building
FHS	Fuel handling system
FOAK	First-of-a-kind
FOCD	Foreign ownership, control, or domination
FPM	Failure probability model
FPM	Failure probability model
FPP	Fire protection program
FRIP	Fuel reactor inlet pipe
FSA	Financial Support Agreement
FSAR	Final Safety Analysis Report
FSC	Flownex Screening Criteria

FSF	Fundamental safety functions
FSSA	Fire Safe Shutdown Analysis
FUCH	Defuel chute
GASM	Gaseous fission product transport model
GASS	Gas analysis and sampling system
GCA	Graphite core assembly
GCC	Graphite core components
GMRS	Ground motion response spectrum
GQDM	Graphite Core Assembly Material Qualification and Design Methodologies
GSI	Generic safety issues
GWNH	GWN Holding, LLC.
HCS	Helium circulator system
HCSC	Helium circulator system controller
HELB	High-energy line break
HFE	Human factors engineering
HGD	Hot gas duct
HGTR	High-temperature gas-cooled reactor
HIPS	Highly integrated protection system
HIS	Human-system interfaces
HLR	High-level requirements
HMI	Human-machine interfaces
HPB	Helium pressure boundary
HPBM	Helium pressure boundary model
HPFWP	High-pressure feedwater pump
HPS	Helium purification system
HRA	Human reliability analysis
HS	Hazards screening
HSCW	Helium services cooling water system
HSF	Helium storage facility
HSHV	Helium service facility HVAC system
HSI	Human-system interface
HSS	Helium service system
HUPs	High-assay low-enriched uranium pebbles
HVAC	Heating, ventilation, and air conditioning
I/O	Input-output
IAEA	International Atomic Energy Agency
IBR	Incorporated by reference
IDP	Integrated decision-making process
IDPP	Integrated decision-making process panel
IE	Initiating event
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
INL	Idaho National Laboratory
IPS	Investment protection system
IPyC	Inner pyrolytic carbon
ISA	International Society of Automation
ISG	Interim Staff Guidance
ISI	Inservice inspection
ISRS	In-structure response spectra
IST	Inservice testing

ITP	Initial test program
IUAT	Inter-unit access tunnel
L&Cs	Limitations and conditions
LBE	Licensing basis event
LCO	Limiting conditions for operation
LD	Large HPB depressurization
LFW	Loss of steam generator feedwater
LHS	Loss of condenser heat sink
LME	Long Mott Energy, LLC.
LMGS	Long Mott Generating Station
LMP	Licensing modernization project
LOIA	Loss of instrument air
LOOP	Loss-of-offsite power
LPF	Loss of primary flow
LRW	Liquid radwaste
LRWH	Liquid radwaste handling system
LSSS	Limiting safety system settings
LTR	Licensing Topical Report
LUPs	Low-enriched uranium pebbles
LWR	Light-water reactors
MANDE	Monitoring and non-destructive examination
MCR	Main control room
MD	Medium HPB depressurization
MEB	Main electrical building
MFLB	Main feedwater line break
MHV	NI module HVAC system
mrem	Millirem
MS	Main steam system
MSLB	Main steam line break
MST	Mechanistic source term
MTRBC	Multi-module total loss of RB cooling water
MWe	Megawatts-electric
MWt	Megawatts-thermal
NAR	Neutron absorber rod
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NGNP	Next Generation Nuclear Plant
NI	Nuclear island
NIAS	Nuclear island auxiliary structure
NICW	Nuclear island cooling water system
NIE	Nuclear island electrical system
NIFI	Nuclear island feedwater isolation
NIFP	Nuclear island fire protection
NIHV	Nuclear island heating, ventilation, and air conditioning system
NIIA	NI instrument air system
NILR	Nuclear island liquid radwaste system
NIPW	Nuclear island process water system
NIS	Nuclear instrumentation system
NISI	Nuclear island steam isolation
non-DBA	Non-design-basis accident

non-LWR	Non-light-water reactor
NRC	U.S. Nuclear Regulatory Commission
NSRST	Non-safety-related with special treatment
NST	No special treatment
NSTF	Natural Convection Shutdown Heat Removal Test Facility
NUMARC	Nuclear Management and Resources Council
OAS	Operator alarm system
OCDC	Owner controlled design criteria
OCDM	Offsite Dose Calculation Manual
ODS	Operator display system
OER	Operating experience review
OL	Operating license
OPS	Operator procedure system
OPyC	Outer pyrolytic carbon
ORNL	Oak Ridge National Laboratory
PBMR	Pebble Bed Modular Reactor
PCDAS	Plant control and data acquisition system
PCP	Process control program
PDC	Principal design criteria
PEMS	Post Event Monitoring System
PFA	Process functional area
PGA	Peak ground acceleration
PITAP	Post-construction inspection, testing, and analysis program
PLOFC	Pressurized loss of forced cooling
PMC	Plant monitoring computer
POS	Plant operating state
PPRS	Primary pressure relief system
PRA	Probabilistic risk assessment
PSAR	Preliminary safety analysis report
PSF	PRA safety functions
QA	Quality assurance
QAP	Quality assurance program
QAPD	Quality assurance program description
QHO	Quantitative health objectives
R&D	Research and development
RAM	Radiation area monitor
RAP	Reliability assurance program
RB	Reactor building
RBC	Reactor building cooling water
RBCW	Reactor building cooling water system
RBHV	Reactor building HVAC system
RC	Radiological consequence
RCA	Radiologically controlled area
RCCS	Reactor cavity cooling system
RCI	Request for confirmation of information
RCS	Reactivity Control System
RCSS	Reactivity control and shutdown system
rem	Roentgen equivalent man
RFDC	Required functional design criteria
RG	Regulatory Guide

RI	Risk integration
RIM	Reliability and integrity management
RIU	Reactor inlet unit
RMS	Radiation Monitoring System
ROST	Remotely operated special tooling
RPI	Rod position and indication system
RPP	Radiation protection program
RPS	Reactor protection system
RPV	Reactor pressure vessel
RSF	Required safety function
RSS	Reserve Shutdown System
RT	Reactor trip
RVI	Reactor vessel internals
RVSS	Reactor vessel support system
RWB	Radwaste building
RWHV	Radwaste building HVAC system
RXST	Reactor support structures
SC	Success criteria
SD	Small HPB depressurization
SDG	Standby diesel generator
SE	Safety evaluation
SFA	Standpipe functional area
SFC	Spent fuel canister
SFFPs	Start-up fuel free pebbles
SFISF	Spent Fuel Intermediate Storage Facility
SFSR	Spent fuel storage rack
SFSS	Spent fuel storage system
SG	Steam generator
SGFS	Steam generator feedwater system
SGPV	Steam generator pressure vessel
SGS	Steam generator system
SGSS	Steam generator steam system
SGST	Steam generator support structures
SGTL	Steam generator tube leak
SGTR	Steam generator tube rupture
SiC	Silicon Carbide
SL	Safety limit
SMS	Seismic monitoring system
SNM	Special nuclear material
SOLM	Solids fission product transport model
SPRA	Seismic PRA
SR	Safety-related
SRDC	Safety-related design criteria
SRM	Staff requirements memorandum
SRV	Safety relief valve
SRWH	Solid radwaste handling system
SSCs	Structures, systems, and components
SSE	Safe shutdown earthquake
SSI	Soil-structure interaction
SSS	Start-up and shutdown system

SST	Shield structure
SUM	Sphere unloading machine
SY	Systems analysis
TDCC	The Dow Chemical Company
TECDOC	Technical document
TEDE	Total effective dose equivalent
TG	Turbine-generator system
THM	Thermodynamics calculation model
TMI	Three Mile Island
TR	Topical report
TRITM	Tritium Production and Transport Model
TS	Technical specifications
TSAM	Transient and safety analysis methodology
TT	Turbine trip
UCO	Uranium Oxycarbide
UFC	Used fuel canister
USI	Unresolved safety issues
USL	Upper safety limit
V&V	Verification and validation
VAC	Volts alternating current
VDC	Volts direct current
VFD	Variable frequency drive
VSOP	Very Superior Old Programs
W	Watts
WSC	Waste Control Specialists, LLC.
XPeRT	X-energy Pebble Reactor Test

1 GENERAL PLANT AND SITE DESCRIPTION AND OVERVIEW OF LICENSING BASIS

1.1 Introduction

This safety evaluation (SE) documents the results of the U.S Nuclear Regulatory Commission (NRC or Commission) staff's technical review of the construction permit (CP) application submitted by Long Mott Energy, LLC., (LME) for the Long Mott Generating Station (LMGS). The LMGS CP application was submitted under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," by letter dated March 31, 2025 (ML25090A057) and proposes to construct an 800 megawatt-thermal (MWth) nuclear power plant comprised of four X Energy, LLC., (X-energy) Xe-100 small modular high-temperature gas-cooled reactors (HTGRs) and associated common facilities in Calhoun County, Texas. The CP application consists of the following parts:

- Part I – General and Financial Information
- Part II – Preliminary Safety Analysis Report
- Part III – Environmental Report
- Part IV – Technical Specifications
- Part V – Non-Applicabilities and Exemptions
- Part VI – Supplemental Information
- Part VII – Security-Related Information to be Withheld per 10 CFR 2.390(d)(1)

The NRC staff published notices in the *Federal Register* (FR) on May 2, 2025; May 9, 2025; May 16, 2025; and May 23, 2025 (90 FR 18874, 90 FR 19733, 90 FR 21080, and 90 FR 22118, respectively) to acknowledge receipt of LME's CP application. The NRC staff conducted an acceptance review for docketing of LME's LMGS CP application and, by letter dated May 12, 2025 (ML25115A247), determined that LME's LMGS CP application was complete and acceptable for docketing. The application was assigned Docket Number (No.) 50-614. A notice of docketing of LME's LMGS CP application was published in the FR on June 10, 2025 (90 FR 24428).

LME submitted Revision 1 of the CP application on September 18, 2026 (ML26261A419) and submitted responses to requests for confirmation of information (RCIs) on August 28, 2026 (ML26239A157) and September 17, 2026 (ML26260C204). Revision 1 of the CP application incorporated the following supplements associated with the safety review to the initial application submitted on March 31, 2025 (ML25090A057):

- Supplement #1 to the preliminary safety analysis report (PSAR), dated September 26, 2025 (ML25269A125).
- Supplement #2 to the PSAR, dated November 20, 2025 (ML25324A306).
- Supplement #3 to the PSAR, dated December 4, 2025 (ML25338A310).

- Supplement #4 to the PSAR, dated December 12, 2025 (ML25346A256).
- Supplement #5 to the PSAR, dated December 17, 2025 (ML25351A147).
- Supplement #6 to the PSAR, dated February 2, 2026 (ML26033A250).
- Supplement #1 to part I of the CP application, dated March 3, 2026 (ML26062A842).
- Supplement #1 to part V of the CP application, dated March 3, 2026 (ML26062A257).

This SE documents the NRC staff's evaluation of the LMGS CP application in accordance with the Atomic Energy Act of 1954, as amended (AEA), and applicable Commission regulations. The NRC staff's SE for the CP application to construct the 10 CFR Part 50 utilization facility is based on information provided by LME in Revision 1 of the CP application and RCI responses. The NRC staff also conducted the following audits: (1) general audit (ML25128A332); (2) probabilistic risk assessment (PRA) audit (ML25156A350); and (3) PSAR chapter 2 and structures audit (ML26048A641) to enable the NRC staff to gain understanding, confirm information, and/or identify information that would require docketing to support the basis for the NRC staff's regulatory decision. The NRC staff summarized the results of these audits in the audit summary issued on October 23, 2026 (ML26293A340). The NRC staff's evaluation of the LMGS preliminary design and analysis was based primarily upon the following 10 CFR requirements:

- 10 CFR 50.30, "Filing of application; oath or affirmation";
- 10 CFR 50.33, "Contents of applications; general information";
- 10 CFR 50.34, "Contents of applications; technical information";
- 10 CFR 50.34(a), "Preliminary safety analysis report";
- 10 CFR 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents - nuclear power reactors";
- 10 CFR 50.35, "Issuance of construction permits";
- 10 CFR 50.40, "Common standards";
- 10 CFR 50.50, "Issuance of licenses and construction permits";
- 10 CFR 50.55, "Conditions of construction permits, early site permits, combined licenses, and manufacturing licenses";
- 10 CFR 50.55a, "Codes and standards";
- 10 CFR 50.150, "Aircraft impact assessment";
- 10 CFR Part 20, "Standards for protection against radiation";

- 10 CFR Part 26, “Fitness for duty programs,” Subpart K, “FFD [fitness for duty] programs for construction”; and

- 10 CFR Part 100, “Reactor site criteria.”

The NRC staff performed its SE using the following primary guidance documents based on LME’s use of the Licensing Modernization Program (LMP):

- Regulatory Guide (RG) 1.233, Revision 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 Nuclear Energy Institute (NEI) 18-04, Revision 1, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” (ML19241A472), with clarifications.

- RG 1.253, Revision 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 NEI 21-07, Revision 1, “Technology-Inclusive Guidance for Non-Light Water Reactors, Safety Analysis Report Content for Applicants Using the NEI 18-04 Methodology,” (ML22060A190), with clarifications and additions.

- Division of Advanced Reactors and Non-Power Production and Utilization Facilities (DANU) Interim Staff Guidance (ISG) DANU-ISG-2022-01, “Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications - Roadmap.”

The LMP guidance in NEI 18-04 defines risk-informed, performance-based, and technology-inclusive processes for the selection of licensing basis events (LBEs); safety classification of structures, systems, and components (SSCs); and the determination of defense-in-depth (DID) adequacy for non-light-water reactors (non-LWRs). NEI 18-04 provides a frequency-consequence target curve that is used to assess events, SSCs, and programmatic controls. LBEs are categorized by the frequency of occurrence, separated into anticipated operational occurrences (AOOs), design-basis events (DBEs), and beyond-design-basis events (BDBEs).

Figure 1.1-1 of this SE provides a high-level flow diagram for the NEI 18-04 process with important components, including the results which are represented in the diagram along with where they are described in the PSAR. Additional information important to the safety basis that is not represented in the figure includes: (1) the site characteristics that determine the design-basis hazard levels and provide atmospheric dispersion information for the consequence analysis in PSAR chapter 2, “Site Characteristics,” (2) the evaluation of DID in PSAR chapter 4, “Integrated Evaluations,” (3) preliminary discussions on aspects primarily pertaining to the operating license (OL) application in PSAR chapter 9, “Control of Routine Plant Radioactive Effluents, Liquid Radwaste, Solid Radwaste, and Plant Contamination,” PSAR chapter 10, “Control of Radioactive Dose,” PSAR chapter 11, “Conduct of Operations,” and PSAR chapter 12, “Initial Test Program,” and (4) the research and development (R&D) program to address remaining safety questions prior to the completion of construction in PSAR chapter 13, “Research and Development Program.”

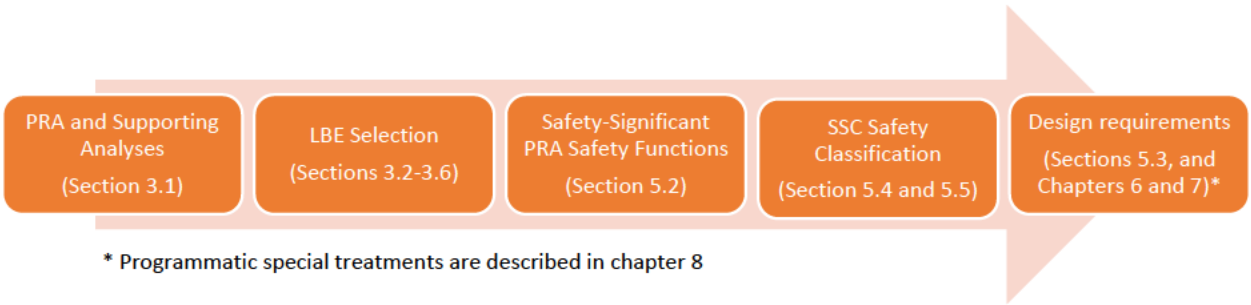

Figure 1.1-1: General flow diagram of the NEI 18-04 process and where results are described in the PSAR

NEI 21-07, Revision 1 describes guidance on the scope and level of detail for specific portions of the first eight chapters of a PSAR that are associated with an LMP-based safety analysis. DANU-ISG-2022-01 provides: (1) a general overview of the information that should be included in a non-LWR application submitted under 10 CFR Part 50 or 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants"; (2) a review roadmap for NRC staff with the principal purpose of ensuring consistency, quality, and uniformity of reviews; and (3) a well-defined base from which the NRC staff can evaluate proposed differences in the scope of reviews (e.g., CP versus OL). RGs 1.233 and 1.253, as well as other key guidance documents, are referenced in DANU-ISG-2022-01.

As appropriate, the NRC staff used additional guidance and codes and standards (e.g., NRC RGs, Institute of Electrical and Electronics Engineers standards, and American National Standards Institute/American Nuclear Society (ANSI/ANS) standards) in its review of the LMGS CP application.

The content of the CP application is evaluated in the following sections of the NRC staff's SE:

- Part I of the CP application, "General and Financial Information," is evaluated in chapter 14 of this SE.
- Part II of the CP application, "Preliminary Safety Analysis Report," consists of 13 chapters that are evaluated in chapters 1 through 13 of this SE. Chapter titles for each chapter of this SE align with the corresponding PSAR chapter titles.
- Part III of the CP application, "Environmental Report," is evaluated in the NRC staff's Environmental Assessment for the LMGS CP application (ML26112A016) and is not addressed in this SE.
- Part IV of the CP application, "Technical Specifications," is evaluation in section 11.7 of this SE.
- Part V of the CP application, "Non-Applicabilities and Exemptions," is evaluated in chapter 15 of this SE.
- Part VI of the CP application, "Supplemental Information," and part VII of the CP application, "Security-Related Information to be Withheld per 10 CFR 2.390(d)(1)" contain information that is considered throughout this SE.

The following additional requirements, not directly addressed by the LMGS CP application, are evaluated as follows:

Mandatory Hearing

On September 22, 2026, the NRC staff held the mandatory hearing for the LMGS CP application (ML26288A056). The NRC staff considered the input received during the mandatory hearing as part of its safety, security, and environmental review. The items identified were addressed during the licensing process or will be addressed as part of future established regulatory processes.

Nuclear Waste Policy Act

The Nuclear Waste Policy Act of 1982, as amended (42 USC § 10101) states that the U.S. Government is responsible for the permanent disposal of high-level radioactive waste and spent nuclear fuel, but the cost of disposal should be the responsibility of the generators and owners of such waste and spent fuel. Guidance for the NRC staff to evaluate compliance with the Nuclear Waste Policy Act is provided in DANU-ISG-2022-01. The applicant's letter dated March 31, 2025 (ML25090A058), submitting the CP application, stated that "LME is in good faith negotiations with the U.S. Department of Energy to enter into a contract for the disposal of high-level waste and nuclear fuel under section 302(b) of the Nuclear Waste Policy Act of 1982, as amended." By letter dated October 2, 2026 (ML26275A165), LME provided documentation from the U.S. Department of Energy (DOE) that LME is actively, and in good faith, negotiating on a contract under section 302(b) of the Nuclear Waste Policy Act. Because LME has provided documentation of good faith negotiations with DOE, the NRC staff finds that LME is in compliance with the Nuclear Waste Policy Act at the CP stage, consistent with DANU-ISG-2022-01.

Ongoing Research and Development

The provisions of 10 CFR 50.34(a)(8) allow for ongoing R&D to confirm the adequacy of the design of SSCs to resolve safety questions prior to the completion of construction. In accordance with 10 CFR 50.34(a)(8), LME identified several R&D activities, which are described in PSAR chapter 13. Chapter 13 of this SE evaluates these activities.

Advisory Committee on Reactor Safeguards Review

To support the Advisory Committee on Reactor Safeguards (ACRS) in providing an independent review and report to the Commission regarding the LMGS CP application, the NRC staff presented the results of its SE to the ACRS subcommittee on September 1, 2026; September 2, 2026; and September 15, 2026, and supported the ACRS full committee meeting on October 7, 2026. After the meetings, the ACRS issued a letter to the Commission (ML26289A141) with their recommendations regarding the LMGS CP application to meet the requirements of 10 CFR 50.58, "Hearings and report of the Advisory Committee on Reactor Safeguards."

Application Availability

Publicly available documents related to the LMGS CP application may be obtained online in the Agencywide Documents Access and Management System (ADAMS) public documents

collection at https://www.nrc.gov/reading-rm/adams.html. To begin the search, select "ADAMS Public Documents" and then select "Begin ADAMS Public Search." For assistance with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209 or via email at PDR.Resource@nrc.gov.

Public documents and correspondence related to this application may be found in ADAMS by searching LMGS's Docket Number 50-614. Portions of the application or correspondence containing sensitive information (e.g., proprietary information) are withheld from public disclosure pursuant to 10 CFR 2.390, "Public inspections, exemptions, requests for withholding."

The NRC Staff's Contact Information

The project manager for this SE was Adrian Muñiz, Senior Project Manager, Office of Advanced Reactors, U.S. NRC. Mr. Muñiz may be contacted regarding this SE at 301-415-4093 or via email at Adrian.Muniz@nrc.gov. Appendix C to this SE provides a list of principal contributors.

1.2 Summary and Conclusions on Principal Safety Considerations

The NRC staff evaluated the descriptions and discussions of the proposed LMGS facility, as described in LME's CP application, as supplemented. Based on its review, the NRC staff makes the following findings:

1. Applicable standards and requirements of the AEA and Commission regulations have been met.

2. The application generally conforms to the applicable guidance for a CP applicant for a commercial non-LWR, with acceptable departures as noted in this SE.

3. Required notifications to other agencies or bodies related to this licensing action have been duly made.

4. Based on the preliminary design of the facility, there is reasonable assurance that the final design will conform to the design basis with adequate margin for safety.

5. There is reasonable assurance that the facility can be constructed in conformity with the permit, the provisions of the AEA, and the Commission's regulations.

6. The NRC staff has evaluated the accident analyses presented by LME in the PSAR and determined that the calculated potential radiation dose consequences outside the LMGS site from postulated accidents are not likely to exceed the dose guidelines of 10 CFR 50.34(a)(1)(ii)(D), taking into consideration the site criteria contained in 10 CFR Part 100." Furthermore, SSCs have been designed to provide for the prevention of accidents and the mitigation of consequences of accidents.

7. Releases of radioactive materials and wastes from the facility are not expected to result in concentrations outside the limits specified by 10 CFR Part 20, Subpart D, "Radiation Dose Limits for Individual Members of the Public," and are as low as is reasonably achievable.

8. The financial information, technical analyses, programs, and organization described in the application, as supplemented, demonstrate that LME is financially and technically qualified to engage in the construction of its proposed facility in accordance with the Commission's regulations.

9. The preliminary emergency plan provides reasonable assurance that LME will be prepared to assess and respond to emergency events.

10. The application presents information at a level of detail that is appropriate for general familiarization and understanding of the proposed facility.

11. The application describes the relationship of specific facility design features to reactor operation.

12. Issuance of the CP will not be inimical to the common defense and security or to the health and safety of the public.

Therefore, the NRC staff finds that, subject to certain conditions, the preliminary design and analysis of the LMGS, as described in the PSAR, is sufficient to meet the applicable regulatory requirements for the issuance of a CP in accordance with 10 CFR 50.35.

The NRC staff notes that there are some remaining areas of uncertainty in the LMGS design and analysis, as discussed throughout this SE. This is acceptable under the requirements of 10 CFR 50.34(a) and 10 CFR 50.35, which state that the expected design and analysis is preliminary and that outstanding safety questions may be resolved through R&D efforts. The NRC staff also notes that, per 10 CFR 50.35(b), a CP does not constitute Commission's approval of any design feature unless specifically requested and incorporated into the permit; no such approval was requested in the CP application for the LMGS. As stated in this SE where applicable, further technical information or design information required to complete the safety analysis can reasonably be left for later consideration in the final safety analysis report (FSAR). The NRC staff notes that resolution of these items is not necessary for the issuance of a CP.

In PSAR chapter 13, LME identified R&D activities necessary to confirm the adequacy of the design of SSCs to resolve safety questions prior to the completion of construction. If an OL application is submitted, the NRC staff will ensure that the R&D activities are appropriately completed pursuant to 10 CFR 50.34(b)(5). Additionally, the NRC staff recommends that the Commission include the permit conditions identified in appendix A of this SE if the CP is issued. These permit conditions ensure that the applicant completes safety-significant design details necessary to protect public health and safety.

Based on the findings documented in this SE, and subject to the permit conditions identified in appendix A of this SE, the NRC staff recommends that the Commission make the following conclusions for the issuance of a CP for the LMGS facility in accordance with 10 CFR 50.35, 10 CFR 50.40, and 10 CFR 50.50:

1. The application meets the standards and requirements of the AEA, and the Commission's regulations, and notifications to other agencies or bodies have been duly made.

2. LME has described the proposed design of the LMGS, including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public.

3. Such further technical or design information as may be required to complete the safety analysis, and which can reasonably be left for later consideration, will be supplied in the FSAR.

4. Safety features or components that require R&D have been described by LME and an R&D program will be conducted that is reasonably designed to resolve any safety questions associated with such features or components.

5. On the basis of the foregoing, there is reasonable assurance that: (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

6. There is reasonable assurance: (i) that the construction of the LMGS will not endanger the health and safety of the public, and (ii) that construction activities can be conducted in compliance with the Commission's regulations.

7. LME is technically qualified to engage in the construction of its proposed facility in accordance with the Commission's regulations.

8. LME is financially qualified to engage in the construction of its proposed facility in accordance with the Commission's regulations.

9. The issuance of a permit for the construction of the LMGS would not be inimical to the common defense and security or to the health and safety of the public.

10. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs, and considering reasonable available alternatives, the issuance of the CP, subject to the conditions for protection of the environment set forth therein, is in accordance with Subpart A, "National Environmental Policy Act - Regulations Implementing Section 102(2)," of 10 CFR Part 51 and all applicable requirements have been satisfied.

1.3 Plant Description, Site Description, Plant Safety Overview, and Definitions

PSAR chapter 1, "General Plant and Site Description and Overview of Licensing Basis," contains the following information:

- PSAR section 1.1 presents information regarding the reactor and its intended use, along with a high-level description of the plant's SSCs.

- PSAR section 1.2 presents a high-level description of the site location.

- PSAR section 1.3 provides a high-level overview of: (1) the selection of LBEs for the LMGS design, (2) required safety functions (RSFs) identified for the LMGS design (including fundamental safety functions (FSFs) and additional RSFs), and (3) the guidance used to evaluate DID adequacy.

- PSAR section 1.4 contains the assessment of the applicability of generic safety issues (GSIs), unresolved safety issues (USIs), and Three Mile Island (TMI) action items. In addition, this section identifies the RGs and consensus codes and standards applied to the LMGS design, and identifies the licensing topical reports (TRs) incorporated by reference into the LMGS PSAR.

- PSAR section 1.5 contains the list of definitions used in the LMGS PSAR.

PSAR sections 1.1 through 1.3 and 1.5 are evaluated in this section and follow the guidance provided in NEI 21-07, as endorsed by RG 1.253.

PSAR section 1.1 provides a description of the type of reactor, the intended use of the reactor, and the overall configuration of the reactor proposed for the LMGS. The reactor model is the Xe-100 HTGR supplied by X-energy. Each reactor is 200 megawatts-thermal (MWt), 80 megawatts-electric (MWe) and is fueled with uranium oxy-carbide (UCO) tri-structural isotropic (TRISO) coated fuel embedded in spherical fuel elements, referred to as pebbles. Each Xe-100 reactor core contains approximately 220,000 pebbles and is cooled by forced circulation of helium through the pebble bed. The heat is transferred from the fuel by the helium to a steam generator (SG) where it is used to generate superheated steam. The steam is used to generate electricity and process steam for Seadrift Operations which is adjacent to the proposed LMGS site.

The LMGS plant includes four Xe-100 reactors and SG pairs located on a single nuclear island (NI) with a shared reactor building foundation and shield structure. Additional systems shared between reactor modules are listed in PSAR section 1.1.3. The sharing of these systems is evaluated in chapters 6 and 7 of this SE through principal design criterion (PDC) 5. PSAR section 1.1.3 also includes a comparison to similar facilities, including Fort St. Vrain, an 842 MWt HTGR that previously operated in the United States.

PSAR section 1.1.4 includes a summary description of plant SSCs, including reactor systems and components, secondary systems and components, significant support systems and components, and major structures. Safety-significant SSCs (i.e., those that are assigned a safety classification of safety-related (SR) or non-safety-related with special treatment (NSRST)) are described in PSAR chapter 6, “Safety-related SSC Criteria and Capabilities,” and PSAR chapter 7, “Non-Safety-Related with Special Treatment Structures, Systems, and Components Criteria and Capabilities,” and evaluated in chapters 6 and 7 of this SE. Liquid radwaste and solid radwaste handling systems are described in PSAR chapter 9 and evaluated in chapter 9 of this SE. However, many non-safety-related with no special treatment (NST) SSCs are described in PSAR section 1.1.4 and evaluated in this chapter of the SE, including:

- reactor systems and components
    - startup fuel free pebbles (SFFPs)
    - startup source
- secondary systems and components

  - main steam system (MS)
  - conventional island condensate and feedwater system (CDFS)
  - condenser system (CDS)
  - turbine-generator system (TG)
  - extraction steam system
  - process steam supply system
  - helium services cooling water system (HSCW)
  - conventional island cooling water system (CICW)
- significant support systems and components
  - remotely operated special tooling (ROST) cart
  - main control room (MCR)
  - plant monitoring computer (PMC)
  - maintenance & diagnostic monitoring system
  - first-of-a-kind data acquisition system
  - conventional island electrical system (CIE)
  - NI liquid drainage system
  - NI instrument air system (NIIA)
  - conventional island compressed and instrument air system (CIIA)
  - nuclear island process water system (NIPW)
- major structures
  - helium storage facility (HSF)
  - access and security building (ASB)
  - fire protection pump house and fire water storage tank structure

In PSAR table 1.4-2, LME states that the LMGS will be in full conformance with RG 1.12, Revision 3, "Nuclear Power Plant Instrumentation for Earthquakes." The applicable regulatory requirement for this review topic is 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," Paragraph IV(a)(4), "Required Seismic Instrumentation." The evaluation of the LMGS seismic monitoring against RG 1.12, Revision 3 and the associated requirements in 10 CFR Part 50, Appendix S is below.

1.3.1 Technical Evaluation

Plant Description

The role of SFFPs and the startup source in the transition from the non-fuel core to the steady-state core is described in PSAR section 6.4.2 and evaluated in section 6.4.2 of this SE. PSAR section 1.1.4.1.2 lists the American Society for Testing and Materials standards used to ensure requirements in the specification are met, including for the graphite feedstock to make sure it is nuclear-grade and for graphite impurities and other key parameters like compressive strength. Testing on wear rate will also be performed to confirm that the pebbles will not break down as they travel through the core. PSAR section 7.3.6 describes the fuel handling system, which also has the capability to remove damaged pebbles from circulation.

PSAR section 6.4.2 describes that the startup source is used in the 1/M approach to critical. PSAR section 3.10.7 provides a preliminary analysis of postulated radiological release from the startup source, which is assumed to be a Cf-252 source with a strength of 400 mCi (14.8 GBq). Final selection of the type and strength of the startup source as well as placement of the startup source will be provided at the OL stage. The NRC staff finds that the basis for classification of

the SFFPs and startup source as NST is acceptable for the CP because it is consistent with the safety analysis documented in PSAR chapter 3, “Methodologies, Analyses, and Licensing Basis Events.”

PSAR section 8.8.2 describes that the CDFS chemical injection subsystem (CDCI) within the CDFS, which is used to provide chemicals needed for corrosion protection and scale build-up prevention to the condensate, feedwater, and steam systems, will be part of implementing the secondary water chemistry objectives of the chemistry program. Applying the chemistry program to the SGs, turbine, and feedwater system ensures the system functions are performed.

The HSCW provides cooling to various NI SSCs and the CICW provides cooling for the feedwater system (CDFW), CDS, MS, and TG. While these systems do not perform a function that mitigates an event, their failure may contribute to the frequency of initiating events such as loss of feedwater. The impact of these systems failing on the overall facility risk will be assessed through the risk-significance evaluations described in the response to RCI-1 dated August 14, 2026 (ML26230A155). The NRC staff finds that the basis for the list of secondary systems and components classified as NST is acceptable for the CP because it is consistent with the current safety analysis documented in PSAR chapter 3.

PSAR section 7.3.1.1 through 7.3.1.3 describes the role of the ROST carts in the transport of spent and used fuel. As stated in PSAR section 7.3.1.3, spent fuel and used fuel storage canisters are designed to survive falls from the ROST cart. Criticality safety analyses involving various canister storage locations and configurations are documented in PSAR section 3.9 and evaluated in section 3.9 of this SE. These analyses are sufficient to bound criticality safety aspects related to the ROST cart for the CP. The NRC staff finds that the basis for the classification of the ROST carts as NST is acceptable for the CP because it is consistent with the safety analysis documented in PSAR chapter 3.

PSAR section 7.3.1.6 describes that the MCR is located on the second floor of the controls and electrical building (CEB), which is classified as NSRST. While the MCR is classified as NST, the CEB is designed to protect operators and the NSRST SSCs in the CEB from eternal hazards such as seismic, flooding, and tornado missiles. PSAR section 5.5.3 describes one NSRST operator action required during some steam generator tube leak (SGTL) events to identify the leaking bundle, isolate it, and restore active cooling through the start-up and shutdown system with the intact bundle. PSAR section 7.3.12 includes additional discussion on how this operator action is performed using NSRST indications, displays, and human-machine interfaces in the distributed control system (DCS). PSAR section 7.3.12 also states that the NI heating, ventilation, and air conditioning (HVAC) system is not required to support this operator action based on the timing of the event and the nuclear island electrical system (NIE) remains available to support the required DCS components because loss of both offsite and backup power concurrent with an SGTL event is below the frequency cutoff for LBEs. The NRC staff concludes that this basis for classification of the MCR, MCR HVAC, and the NIE supporting this function meets the requirements for the issuance of a CP because the preliminary design addresses the safety aspects for the scope of LBEs in PSAR chapter 3.

The PMC provides operators with the capability to monitor the plant from outside the MCR, consistent with the requirements of PDC 19. The PMC also provides radiation monitoring and seismic monitoring through dedicated subsystems, consistent with the requirements of PDC 13. The radiation monitoring system (RMS) provides instrumentation, monitoring, and alarms in fuel

storage and radioactive waste systems consistent with PDC 63. In addition, the RMS provides airborne monitoring in the reactor building and on effluent discharge paths consistent with PDC 64. The NRC staff determined that the preliminary design of the PMC supports compliance with 10 CFR 50.34(f)(2)(iv), 10 CFR 50.34(f)(2)(xviii), 10 CFR 50.34(f)(2)(xix), 10 CFR 50.34(f)(2)(xxvii), and is consistent with RG 1.97, and RG 1.166 to the extent described in PSAR table 1.4-2.

The PMC also includes the seismic monitoring system (SMS). The seismic instrumentation in the SMS consists of 11 in-structure monitoring instruments, and 2 free-field monitoring instruments that are located at foundation depth and at a sufficient distance away from major structures. The NRC staff determined that the information provided on the number and locations of seismic information is satisfactory because the applicant follows the recommendations of RG 1.12, Revision 3 and therefore meets the regulatory requirements of 10 CFR Part 50, Appendix S, for providing suitable instrumentation so that the seismic response of nuclear power plant features important to safety can be evaluated promptly after an earthquake.

Like the HSCW and CICW; the NIIA, CIIA, and NIPW do not perform a function that mitigates an event, but their failure may contribute to the frequency of various initiating events based on their support of NSRST or SR systems functions. The impact of these systems failing on the overall facility risk will be assessed through the risk-significance evaluations described in the response to RCI-1. The NRC staff concludes that the basis for the list of significant support systems and components classified as NST meets the requirements for the issuance of a CP because it is consistent with the current safety analysis documented in PSAR chapter 3.

The HSF provides storage and processing of helium used in the reactor system and is designed to prevent collapse. The NRC staff concludes that the basis for the list of major structures classified as NST meets the requirements for the issuance of a CP because it is consistent with the current safety analysis documented in PSAR chapter 3.

Consistent with the LMP process, LME will continue to assess the classifications for these systems as the design and the safety analysis mature and as additional events are added to the PRA as the design transitions from preliminary to final between the CP and OL.

Site Description

PSAR section 1.2 and PSAR figures 1.1-3 and 1.2-1 provide a high-level site description. Detailed information on the site can be found in PSAR chapter 2, and the NRC staff's evaluation of the site information can be found in chapter 2 of this SE.

Plant Safety Overview

As described above, LME uses the risk-informed and performance-based LMP methodology established in NEI 18-04, as endorsed with clarifications in RG 1.233, as the safety analysis methodology for LMGS. The evaluations for LME's implementation of this methodology are primarily contained in SE section 3.2 for LBE selection, SE section 5.1 for SSC classification, and SE section 4.2 for the DID evaluation.

PSAR section 1.3 contains an overview of the FSFs of retaining radionuclides, controlling reactivity, controlling heat removal, controlling water/steam ingress, and maintaining geometry.

These FSFs align with the RSFs described in PSAR sections 5.2 and 5.4, and are evaluated in sections 5.2 and 5.4 of this SE.

Definitions

PSAR table 1.5-1 contains definitions used throughout the PSAR. These definitions are primarily from NEI 18-04, and the NRC staff concludes that the definitions in the table are consistent with the guidance.

1.3.2 Conclusions

Based on the technical evaluation above, the NRC staff concludes that the information provided in PSAR sections 1.1 through 1.3 and 1.5 meets the requirements for the issuance of a CP and is consistent with the content-of-application requirements in NEI 21-07 as endorsed by RG 1.253.

1.4 Regulatory Analysis

PSAR section 1.4 contains the following:

- PSAR section 1.4.1 and associated PSAR table 1.4-1 assess the applicability of GSIs, USIs, and TMI action items.
- PSAR section 1.4.2 and associated PSAR table 1.4-2 identify regulatory guides applied to the LMGS design.
- PSAR section 1.4.3 and associated PSAR table 1.4-3 identify consensus codes and standards applied to the LMGS design.
- PSAR section 1.4.4 and associated PSAR table 1.4-4 identify the licensing TRs that are incorporated by reference (IBR) into the LMGS PSAR.
- PSAR section 1.4.5 states that: “The LMGS plant is a single nuclear power plant incorporating a small multi-module reactor design. There are no existing or planned operating nuclear power plants co-located on the LMGS site to which LMGS construction could pose a hazard. Thus, the LMGS site is not a multi-unit site as contemplated by 10 CFR 50.34(a)(11). As such, the construction hazards evaluation and managerial and administrative controls required by 10 CFR 50.34(a)(11) are not germane to the LMGS.”

Regarding the consideration of USIs and GSIs in PSAR section 1.4.1, staff requirements memorandum (SRM)-SECY-15-0002 approved SECY-15-0002, “Updating New Reactor Licensing Policies, Rules, and Guidance,” Enclosure 1, “Improving Alignment Between New Reactor Licensing Processes,” (ML13277A420) which states:

> *While Part 50 does not contain regulations identical to 10 CFR 52.47(a)(21), 52.47(a)(22), or 52.79(a)(20), legal precedents establish that applicants under 10 CFR Part 50 must demonstrate the resolution of USIs and GSIs prior to issuance of an OL. In accordance with the requirements in 10 CFR 50.34 and*

10 CFR 50.35, a CP applicant must demonstrate by a sufficient level of information that all safety issues are capable of being resolved or otherwise addressed prior to issuance of an operating license. Specifically, the ruling of the Atomic Safety and Licensing Appeal Board (ALAB) in Long Island Lighting Company (Shoreham Nuclear Power Station Unit 1) ALAB-788, 20 NRC 1102 (1984) provides the basis for this position.

DANU-ISG-2022-01, Appendix B, "Analysis of Applicability of NRC Regulations to Non-Light-Water Power Reactors" clarifies that: "Requirements under 10 CFR 50.34(f) (i.e., TMI requirements) are only applicable for 10 CFR Part 52 applicants," but footnote 6 of Appendix B clarifies the Commission's direction for new reactor applications to address technically relevant TMI-related items under 10 CFR 50.34(f).

1.4.1 Technical Evaluation

Evaluation of USIs and GSIs

The NRC staff reviewed the information regarding GSIs and USIs presented in PSAR section 1.4.1 to assess whether the information is sufficient information to demonstrate that applicable GSI and USIs can be resolved prior to issuance of an OL.

PSAR section 1.4.1 states that:

- "[TMI] Action Items identified in NUREG-0933, Appendix B, Section 1, are required to be addressed by 10 CFR 50.34(f). Table 1.4-4 [sic] provides a summary of the evaluation of the applicability of these requirements to the design."

- "The remaining generic issues that are identified in NUREG-0933, Appendix B, were also evaluated. There were no items that were identified to be applicable to the design of LMGS."

The NRC staff audited LME's regulatory analysis of GSIs, USIs, and TMI action items to support its evaluation of these statements. This regulatory analysis identified a list of GSIs and USIs that were potentially applicable to LMGS. Based on the NRC staff's review of the potentially applicable GSIs and USIs, the NRC staff confirmed that the GSIs and USIs were not applicable to the LMGS design or were addressed through other regulatory requirements that were evaluated as part of the CP application review.

Evaluation of TMI Action Items under 10 CFR 50.34(f)

Paragraph 50.34(f), "Additional TMI-related requirements," of 10 CFR contains requirements with which compliance must be demonstrated by specifically listed 10 CFR Part 50 applicants and by 10 CFR Part 52 applicants to the extent that the portions of the requirements are technically relevant. PSAR table 1.4-1 provides a summary of the applicant's evaluation of the technical relevance of the portions of 10 CFR 50.34(f) to the LMGS design. Further, for each portion of 10 CFR 50.34(f) identified as technically relevant, the table states whether the LMGS design fully complies with, or partially complies with, that portion of the regulation and which PSAR section(s) discusses this compliance. As identified in the NRC's Advanced Reactor Content of Application Project, which includes DANU-ISG-2022-01, compliance with

10 CFR 50.34(f) is required for 10 CFR Part 52 applicants but is not required for 10 CFR Part 50 applicants that are not specifically listed in the regulation. However, the Commission confirmed in SRM-SECY-15-0002, "Proposed Updates of Licensing Policies, Rules, and Guidance for Future New Reactor Applications" (ML15266A023), that its direction regarding 10 CFR Part 52 applicants applies to new 10 CFR Part 50 applicants. Additionally, consistent with this Commission direction, on September 24, 2026, the NRC published the proposed rule "Regulatory Enhancements for Reactor Licensing, Decommissioning, and Operational Oversight" (RIN 3150-AL45) in the FR (91 FR 60702), which would amend 10 CFR 50.34(f) so that each applicant for a power reactor CP or OL under 10 CFR Part 50 must, like 10 CFR Part 52 applicants, demonstrate compliance with the technically relevant portions of 10 CFR 50.34(f). Therefore, because the applicant identified portions of 10 CFR 50.34(f) that are technically relevant to the LMGS design, and because the regulations may require compliance with technically relevant portions of 10 CFR 50.34(f) at the time of the submittal of an OL application for the LMGS, the NRC staff reviewed the compliance of the LMGS design with those portions of 10 CFR 50.34(f). This review is summarized, as applicable, in the relevant sections of this SE.

Additionally, PSAR table 1.4-1 provides a summary of the applicant's evaluation of 10 CFR 50.34(f) applicability to the LMGS design. SE table 1.3-1 below provides the NRC staff's evaluation of 10 CFR 50.34(f) applicability items against DANU-ISG-2022-01, appendix B, table 4, or the SE section where the NRC staff's evaluation of the item can be found, as applicable to the LMGS design.

Table 1.4-1. NRC Staff's Evaluation of the Applicability of TMI Requirements for the LMGS

DANU-ISG-2022-01, Appendix B, Table 4			PSAR Table 1.4-1		NRC Staff's Evaluation
TMI Requirement	**Topic**	**Applicability to non—LWRs**	**Compliance Status**	**PSAR Section**	
§ 50.34(f)(1)(i)	PRA to seek improvements in reliability of heat removal systems	Yes	Full Compliance	3.1.1	PSAR section 3.1.1 provides a description of the LMGS PRA, which is used to identify improvements in the reliability of core and containment heat removal systems. The NRC staff's evaluation of the PRA is in section 3.1.1 of this SE.
§ 50.34(f)(1)(iii)	Reactor coolant pump seal damage	Yes (entry condition: only for reactor designs that have a coolant pump with seals that retain inventory credited for core cooling)	Not Applicable (N/A)	N/A	The NRC staff agrees with LMEs basis for this item not being applicable.
§ 50.34(f)(2)(i)	Control room simulator	Yes (noting that the discussion of small break loss of coolant accidents may not be technically relevant to some non-LWR designs)	Full Compliance	11.3	PSAR section 11.3 discusses the simulator capability that will model the control room to be submitted prior to submitting the OL application. The NRC staff's evaluation is in section 11.3 of this SE.
§ 50.34(f)(2)(ii)	Plant procedure improvement program	Yes	Full Compliance	8.7, 11.5	PSAR sections 8.7 and 11.5 discuss expanding efforts to improve plant procedures related to human factors engineering, emergency procedures, reliability analyses, crisis management and operator training efforts which are to begin during construction and continue throughout operation. The NRC staff's evaluation is in sections 8.7 and 11.5 of this SE.

DANU-ISG-2022-01, Appendix B, Table 4			PSAR Table 1.4-1		NRC Staff's Evaluation
TMI Requirement	**Topic**	**Applicability to non—LWRs**	**Compliance Status**	**PSAR Section**	
§ 50.34(f)(2)(iii)	Control room human factors	Yes	Full Compliance	1.1.4, 8.7	PSAR sections 1.1.4 and 8.7 provide a description of a preliminary control room design that reflects current human factor principles. The NRC staff's evaluation is in section 8.7 of this SE.
§ 50.34(f)(2)(iv)	Safety parameter display system	Yes	Full Compliance	1.1.4, 6.4.13, 8.7	PSAR section 1.1.4 lists monitoring systems that provide a plant safety parameter display console using the PMC, which is designed to support compliance with 10 CFR 50.34(f)(2)(iv). PSAR section 6.4.13 discusses the reactor protection system (RPS), with the human factors engineering aspect discussed in PSAR section 8.7. The NRC staff's evaluations of the PMS, RPS, and human factors engineering are in sections 1.3.1, 6.4.13, and 8.7 of this SE, respectively.
§ 50.34(f)(2)(v)	Automatic indication of status of safety systems	Yes	Full Compliance	1.1.4, 6.4.13, 8.7	PSAR section 1.1.4 lists monitoring systems that provide a plant safety parameter display console using the PMC, which is designed to support compliance with 10 CFR 50.34(f)(2)(v). PSAR section 6.4.13 discusses the RPS, with the human factors engineering aspect discussed in PSAR section 8.7. The NRC staff's evaluations of the PMC, RPS, and human factors engineering are in sections 1.3.1, 6.4.13, and 8.7 of this SE, respectively.

DANU-ISG-2022-01, Appendix B, Table 4			PSAR Table 1.4-1		NRC Staff's Evaluation
TMI Requirement	**Topic**	**Applicability to non—LWRs**	**Compliance Status**	**PSAR Section**	
§ 50.34(f)(2)(vi)	High point venting of reactor coolant system (RCS)	Yes (entry condition: only if reactor coolant flow is credited for core cooling and coolant flow can be impeded by noncondensible gases)	N/A	6.4.11, 7.3.5	PSAR section 6.4.11 discusses the reactor cavity cooling system (RCCS) and PSAR section 7.3.5 discusses the helium circulator system (HCS). The NRC staff agrees with LME's basis for this item not being applicable because venting of gases is not required to satisfy core cooling for RCCS and HCS, as evaluated in sections 6.4.11 and 7.3.5 of this SE.
§ 50.34(f)(2)(vii)	Radiation shielding design review	Yes	Full Compliance	6.4, 7.3, 9.2, 9.3	PSAR sections 6.4 and 7.3 describe SR and NSRST systems, respectively. PSAR sections 9.2 and 9.3 describe the liquid radwaste and solid waste management as well as the high-level information on radiation and shielding design of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials. The NRC staff finds that the preliminary design description demonstrates that these systems permit adequate access to important areas and that these systems protect safety equipment from the radiation environment.
§ 50.34(f)(2)(viii)	Post-accident sampling	Yes	Full Compliance	7.3.7	PSAR section 7.3.7 describes the helium service system (HSS) which provides the capability to promptly obtain and analyze samples from the helium pressure boundary (HPB) without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities. The NRC staff determined that this sampling

DANU-ISG-2022-01, Appendix B, Table 4			PSAR Table 1.4-1		NRC Staff's Evaluation
TMI Requirement	**Topic**	**Applicability to non—LWRs**	**Compliance Status**	**PSAR Section**	
					location covers the RCS and containment discussed in this TMI item because the functional containment is primarily provided by the fuel, and failure of the fuel would lead to increased radionuclides in the HPB. The NRC staff's evaluation of the HSS is discussed in section 7.3.7 of this SE.
§ 50.34(f)(2)(x)	Relief and safety valves	Yes (entry condition: only if RCS has relief valves and failure of these valves would lead to core cooling challenges)	N/A	N/A	The NRC staff agrees with LME's basis for this item not being applicable.
§ 50.34(f)(2)(xi)	Relief and safety valves	Yes (entry condition: only if RCS has relief valves and failure of these valves would lead to core cooling challenges)	N/A	N/A	The NRC staff agrees with LME's basis for this item not being applicable.
§ 50.34(f)(2)(xiv)	Containment isolation	Yes (entry condition: only for designs that use a traditional containment rather than a functional containment approach)	N/A	N/A	The NRC staff agrees with LME's basis for this item not being applicable.

DANU-ISG-2022-01, Appendix B, Table 4			PSAR Table 1.4-1		NRC Staff's Evaluation
TMI Requirement	**Topic**	**Applicability to non—LWRs**	**Compliance Status**	**PSAR Section**	
§ 50.34(f)(2)(xv)	Containment purging	Yes (entry condition: only for designs that use a traditional containment rather than a functional containment approach)	N/A	N/A	The NRC staff agrees with LME's basis for this item not being applicable.
§ 50.34(f)(2)(xvii)	Control room instrumentation for containment functions	Yes (entry condition: only for designs that use a traditional containment rather than a functional containment approach)	Partially Applicable	8.7, 9.1, 9.4	The NRC staff agrees that the aspects related to containment pressure, water level, and hydrogen containment monitoring are not technically relevant based on the LMGS design. While hydrogen can be generated in tube leak or rupture events, the reactor building and shield structure are not a leak-tight containment and the hazard is addressed through design to ensure that a flammable mixture will not be reached. PSAR sections 9.4 and 9.1 describe monitoring of direct radiation levels and gaseous effluents at potential accident release points, and also describe onsite capability to analyze and measure those samples as part of human-system interface (HSI) discussed in PSAR section 8.7. These systems support meeting 10 CFR 50.34(f)(2)(xvii). The NRC staff's evaluation of the HSI, gaseous effluents, and radiation monitoring capabilities are in sections 8.7, 9.1 and 9.4 of this SE, respectively.
§ 50.34(f)(2)(xviii)	Coolant instrumentation	Yes	Full Compliance	6.4.13, 8.7	PSAR section 6.4.13 describes the instruments in the control room that provide

DANU-ISG-2022-01, Appendix B, Table 4			PSAR Table 1.4-1		NRC Staff's Evaluation
TMI Requirement	**Topic**	**Applicability to non—LWRs**	**Compliance Status**	**PSAR Section**	
					an unambiguous indication of core cooling using plant control and data acquisition system designed to support compliance with 10 CFR 50.34(f)(2)(xviii). PSAR section 8.7 describes the related human factors engineering system which is also designed to support meeting the requirements of 10 CFR 50.34(f)(2)(xviii). The NRC staff's evaluation of the monitoring systems and the human factors engineering is in sections 6.4.13 and 8.7 of this SE.
§ 50.34(f)(2)(xix)	Post-accident monitoring	Yes	Full Compliance	1.1.4, 8.7, 6.4.13	PSAR section 8.7 describes the preliminary HSI design including instrumentation adequate for monitoring plant conditions following any LBE. These monitored parameters include variables that are processed by the post event monitoring system (PEMS), described in PSAR section 1.1.4, or RPS, described in PSAR section 6.4.13, and are monitored on the safety parameter displays in the MCR. The NRC staff's evaluation of the HSI design, PEMS, and RPS are discussed in sections 8.7, 1.1.4, and 6.4.13 of this SE, respectively.

DANU-ISG-2022-01, Appendix B, Table 4			PSAR Table 1.4-1		NRC Staff's Evaluation
TMI Requirement	**Topic**	**Applicability to non—LWRs**	**Compliance Status**	**PSAR Section**	
§ 50.34(f)(2)(xxvi)	Leakage control outside containment	Yes (entry condition: only for designs that have SSCs capable of circulating radioactive materials resulting from an accident outside of qualified barrier(s) to radioactive release)	N/A	N/A	The NRC staff agrees with LME's basis for this item not being applicable.
§ 50.34(f)(2)(xxvii)	In-plant Radiation Monitoring	Yes	Full Compliance	8.7, 10.1	PSAR section 8.7 describes the HSI design including a description of monitoring of in-plant radiation and airborne radioactivity under normal operating and LBE conditions, as required in 10 CFR 50.34(f)(2)(xxvii). PSAR section 10.1 describes the radiation protection program which provides further discussion on in-plant radiation monitoring. The NRC staff's evaluation of the HSI is in section 8.7 of this SE and the evaluation of the radiation protection program is in section 10.1 of this SE.
§ 50.34(f)(2)(xxviii)	Preclude control room habitability issues during accidents	Yes	Full Compliance	10.2	PSAR section 10.2 provides discussion on the as low as reasonably achievable (ALARA) program. PSAR section 10.3 provides discussion on design considerations, such as those described in RG 8.8 which may include shielding or HVAC design, and will be provided in the OL application to demonstrate control room habitability. PSAR section 3.2.2.2 states that control room doses for AOOs and

DANU-ISG-2022-01, Appendix B, Table 4			PSAR Table 1.4-1		NRC Staff's Evaluation
TMI Requirement	**Topic**	**Applicability to non—LWRs**	**Compliance Status**	**PSAR Section**	
					DBEs will remain below 5 rems. With no operator actions needed during BDBEs and design-basis accidents, control room dose was not calculated for these events. The NRC staff determined that this is acceptable for the CP stage based on the LBEs and the limited scope of operator actions. The NRC staff's evaluation of the ALARA program is in section 10.2 of this SE and the NRC staff's evaluation of design considerations applicable to 10 CFR 50.34(f)(2)(xxviii) are in section 10.3.1 of this SE.
§ 50.34(f)(3)(i)	Industry experience	Yes	Full Compliance	8.7, 10.1, 11.6, 12.1.4	PSAR sections 8.7, 10.1, 11.6 and 12.1.4 discuss programs designed to provide administrative procedures for evaluating operating design and construction experience and for ensuring that any applicable industry experience will be shared with those designing and constructing the plant. The NRC staff's evaluation of the human factors engineering program, the radiation protection program, the security and fitness for duty program, and the initial test program are provided in sections 8.7, 10.1, 11.6 and 12.1 of this SE, respectively.
§ 50.34(f)(3)(ii)	Quality assurance (QA) list includes all SSCs important to safety	Yes	Full Compliance	8.1	PSAR section 8.1 provides a description of the QA program (QAP) applied to all SSCs important to safety, with the full scope of the QAP applied to SR SSCs and a graded approach applied to NSRST SSCs,

DANU-ISG-2022-01, Appendix B, Table 4			PSAR Table 1.4-1		NRC Staff's Evaluation
TMI Requirement	**Topic**	**Applicability to non—LWRs**	**Compliance Status**	**PSAR Section**	
					consistent with the LMP methodology. The NRC staff's evaluation of the QAP is in section 8.1 of this SE.
§ 50.34(f)(3)(iii)	QAP	Yes	Full Compliance	8.1	PSAR section 8.1 discusses the LMGS QAP which the NRC staff determined meets the criteria to support compliance with requirements A through G of 10 CFR 50.34(f)(3)(iii). The NRC staff's evaluation of the QAP is in section 8.1 of this SE.
§ 50.34(f)(3)(iv)	Dedicated containment penetrations	Yes (entry condition: only for designs that use a traditional containment rather than a functional containment approach)	N/A	N/A	The NRC staff agrees with LME's basis for this item not being applicable.
§ 50.34(f)(3)(vi)	Containment	Yes (entry condition: only for designs with external hydrogen mitigation systems with a traditional containment)	N/A	N/A	The NRC staff agrees with LME's basis for this item not being applicable.
§ 50.34(f)(3)(vii)	Management plan for design and construction activities	Yes	Full Compliance	11.2	PSAR section 11.2 provides a description of the LMGS organization to support compliance with requirements A through E of 10 CFR 50.34(f)(3)(vii). The NRC staff's evaluation of the organization for LMGS is provided in section 11.2 of this SE.

TRs that are IBR into the LMGS PSAR

PSAR section 1.4.4 and associated table 1.4-4 list NRC-approved TRs that are IBR into the LMGS PSAR. Several of the TRs that are IBR have limitations and conditions (L&Cs) associated with their use. PSAR tables 1.4-5 through 1.4-15 present the L&Cs associated with the TRs along with the methods of addressing each L&C.

- The NRC staff reviewed PSAR table 1.4-5 which addresses the L&Cs associated with "Xe-100 Licensing Topical Report TRISO-X Pebble Fuel Qualification Methodology," Revision 3 (ML23160A294(Public)/ML23160A293(Non-Public)). Table 6.4-2 of this SE presents the NRC staff's evaluation of how LME addressed the L&Cs associated with this TR.

- The NRC staff reviewed PSAR table 1.4-6 which addresses the L&Cs associated with "Xe-100 Licensing Topical Report Risk-Informed Performance-Based Licensing Basis: X-energy's Approach to NEI 18-04 Implementation," Revision 2 (ML22308A197). To address limitation 1 of this TR, which states that any applicant using this SE will need to address compliance with pertinent regulations and request exemptions as needed, LME stated that exemptions to pertinent regulations are addressed in part V of the CP application. LME's proposed exemptions are evaluated in chapter 15 of this SE.

  Based on this, the NRC staff determined that LME adequately addressed the limitation associated with the TR.

- The NRC staff reviewed "Xe-100 Licensing Topical Report: Control Room Staffing Analysis Methodology and Associated Implementation Plans," Revision 2, (ML23258A243) and confirms that there are no applicable L&Cs pertaining to the use of this TR for the CP application.

- The NRC staff reviewed PSAR table 1.4-7 which addresses the L&Cs associated with "Xe-100 Licensing Topical Report Principal Design Criteria," Revision 3, (ML24319A155). The NRC staff's evaluation of L&Cs associated with this TR are presented in section 5.3.1 of this SE.

- The NRC staff reviewed PSAR table 1.4-8, which addresses the L&Cs associated with "Long Mott Energy, LLC Quality Assurance Program Description," (hereafter referred to as LME QAPD) (ML25269A117). LME confirmed that it is complying with the two limitations in the SE for the LME QAPD and committed to addressing limitation 2 in the future if LME decides to procure from a manufacturing licensee. The NRC staff determined that LME adequately addressed the LME QAPD SE limitations acceptable for the CP application stage.

- The NRC staff reviewed PSAR table 1.4-9 which addresses the L&Cs associated with "Xe-100 Licensing Topical Report Atmospheric Dispersion and Dose Calculation Methodology," Revision 2 (ML24302A299). The NRC staff's evaluation of L&Cs associated with this TR is presented below:

  1. To address limitation 1, the applicant provided the assumptions used to ensure that the exclusion area boundary dose is bounding for any 2-hour period by: (1) ensuring the use

of conservative breathing rates during the first 24 hours and following RG 1.183, Revision 0 for 1-30 days, (2) using conservative atmospheric dispersion factors, and (3) not including assumptions for decay for each event resulting in a release. Given that releases are dominated by early phases of the accident and no expectation for a delayed release, the NRC staff understood that the proposed assumptions appear conservative for determining a bounding 2-hour dose given the use of conservative breathing rates, maximum dispersion value, no decay assumed, and releases which would primarily occur in the first 24 hours.

2. To address condition 1, the applicant states in PSAR table 1.4-9 that it will install meteorological instrumentation and will collect site representative data in accordance with RG 1.23. In addition, the applicant states that a site-specific atmospheric dispersion analysis using this data will be performed for an OL application. This method includes confirmation that Xe-100 analysis results using the CP's generic dispersion factors will either be shown to be conservatively bounding or LME will update the LMGS licensing basis with site-specific dispersion factors.

The NRC staff determined that LME adequately addressed the L&Cs for this TR for the preliminary design.

- NRC staff reviewed PSAR table 1.4-10 which addresses the L&Cs associated with "Xe-100 Licensing Topical Report GOTHIC and Flownex Analysis Codes Qualification," Revision 3 (ML25262A060). The NRC staff's evaluation of L&Cs associated with this TR is presented in section 3.1.3.1 of this SE.

- NRC staff reviewed PSAR table 1.4-11 which addresses the L&Cs associated with "Xe-100 Licensing Topical Report Transient and Safety Analysis Methodology," Revision 2 (ML25262A031). The NRC staff's evaluation of L&Cs associated with this TR is presented in section 3.1.2.1 of this SE.

  The NRC staff's evaluation for this TR discusses the 3 limitations and 14 conditions associated with this TR. The NRC staff determined that LME has used the evaluation model described in the TR consistent with the limitations identified in the TR SE. The TR conditions are primarily related to the completion of activities identified in Steps 1 through 20 of the Evaluation Model Development and Assessment Process (EMDAP) in RG 1.203. The applicant has summarized its plans to satisfy these conditions and to complete the EMDAP steps for the evaluation model in PSAR table 1.4-11. The NRC staff determined that LME has provided an acceptable plan that is consistent with RG 1.203.

- The NRC staff reviewed "Xe-100 Licensing Topical Report Mechanistic Source Term Approach," Revision 3 (ML25262A107) and confirms that there are no applicable L&Cs pertaining to the use of this TR for the CP application.

- The NRC staff reviewed PSAR table 1.4-12 which addresses the L&Cs associated with "Xe-100 Licensing Topical Report Reactor Core Design Methods and Analysis," Revision 2 (ML25262A123). The NRC staff's evaluation of L&Cs associated with this TR is presented below:

1. Limitation 1 of the TR states that application of the TR is limited to preliminary safety analysis of the Xe-100 design. This limitation is met since LME is applying the TR to support the safety analysis of the LMGS CP application.

2. Condition 1 of the TR states that a CP applicant referencing the TR must confirm or otherwise justify the regulatory basis for assessments that use the methods in the TR. LME indicated, and the NRC staff verified, that the regulatory basis supporting the methods used in the TR is consistent with the PSAR.

Therefore, the NRC staff determined that LME adequately addressed the L&Cs for this TR.

- The NRC staff reviewed PSAR table 1.4-13 which addresses the seven L&Cs associated with "Xe-100 Licensing Topical Report Plume Exposure Pathway Emergency Planning Zone Sizing Methodology," Revision 1 (ML26075E856). The NRC staff's review of these L&Cs is presented in section 11.4 of this SE.

- The NRC staff reviewed PSAR table 1.4-14 which addresses the L&Cs associated with "Xe-100 Licensing Topical Report Graphite Core Assembly Material Qualification and Design Methodologies," Revision 1 (ML26114A162). The SE for this TR has imposed 25 conditions pertaining to referencing the TR for a CP application. In PSAR table 1.4-14, LME proposed a plan to meet each of the 25 conditions as part of the R&D program described in PSAR chapter 13.

  The NRC staff finds that LME's proposed plan to address the L&Cs using testing, analytical methodology development, or a combination of both, where applicable, will be sufficient to resolve the associated safety questions. The NRC staff also finds that adding the work associated with meeting these L&Cs to the R&D program and addressing it prior to the completion of the OL application, is appropriate. The acceptability is based on the preliminary design information in the CP providing reasonable assurance that the graphite core assembly will perform its intended functions (RSFs and PRA safety functions (PSFs)) as described in PSAR section 6.4.4, and that any unresolved questions will be addressed as part of R&D prior to the completion of construction.

- The NRC staff reviewed PSAR table 1.4-15 which addresses the L&Cs associated with "NuScale Power Licensing Topical Report Design of the Highly Integrated Protection Platform," Revision 2 (ML17256A892 (Public)/ML17256A894 (Non-Public)). The NRC staff's evaluation of L&Cs associated with this TR is presented below:

  - Condition 1 of the SE to this TR requires applicants and licensees to satisfactorily address the set of application-specific action items (ASAIs) for use of the highly integrated protection system platform. To address the limitations, LME has committed in PSAR section 6.4 to addressing the ASAIs and to supplementing the TR as part of an OL application. The NRC staff finds this approach reasonable given the preliminary nature of the LMGS design.

Therefore, the NRC staff determined that LME deferring the need to address the limitations associated with this TR with a commitment to address the ASAIs as part of the OL application is acceptable.

Based on the discussion provided above, the NRC staff determined that the applicant adequately addressed the L&Cs for TRs that are IBR into the LMGS PSAR.

Evaluation Under 10 CFR 50.34(a)(11) for Multiunit Sites

PSAR section 1.4.5 indicates that the requirements under 10 CFR 50.34(a)(11) are not applicable to the LMGS CP application because the LMGS is a single nuclear power plant with four reactor modules and there are no existing or planned nuclear power plants co-located on the LMGS site. The NRC staff agrees that since 10 CFR 50.34(a)(11) is only applicable to multiunit sites, that the regulation is therefore not applicable to the single unit nuclear power plant proposed for the LMGS. Interactions and safety considerations for sharing of systems between modules, within the plant, are assessed through the evaluation of PDC 5 located throughout chapter 6 and 7 of this SE.

1.4.2 Conclusions

Based on the information provided in PSAR chapter 1:

- The NRC staff concludes that the content in PSAR sections 1.1 through 1.3 is consistent with NEI 21-07 as endorsed by RG 1.253.

- The NRC staff concludes that the classification of NST systems as described in PSAR section 1.1.4 are acceptable for the CP based on the preliminary design and safety analysis documented in PSAR chapter 3.

- The NRC staff concludes that LME adequately addressed the USIs and medium- and high-priority GSIs in NUREG-0933.

- The NRC staff concludes that the applicant's identification of applicable requirements under 10 CFR 50.34(f) is acceptable.

- The NRC staff concludes that the applicant adequately addressed the L&Cs for TRs that are IBR into the LMGS PSAR.

- The NRC staff concludes that 10 CFR 50.34(a)(11) is not applicable to the LMGS.

2 SITE CHARACTERISTICS

This chapter of the safety evaluation (SE) describes the U.S. Nuclear Regulatory Commission (NRC) staff's review and evaluation of the Long Mott Generating Station (LMGS) preliminary safety analysis report (PSAR) chapter 2, "Site Characteristics," which contains preliminary information on site characteristics. The purpose of evaluating the site characteristics for a proposed reactor facility is to determine whether the site selected is suitable to construct and operate the proposed facility. Site characteristics evaluated include geography and demography; nearby industrial, transportation, and military facilities; meteorology; hydrology; and geology, seismology, and geotechnical engineering.

2.1 Geography and Demography

PSAR section 2.1 discusses site geography and demography and is divided into three subsections:

- Section 2.1.1, Site Location and Description
- Section 2.1.2, Exclusion Area Authority and Control
- Section 2.1.3, Population Distribution

The applicable regulatory requirements related to geography and demography are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a)(1)(ii), "Contents of Application";
- 10 CFR 100.3, "Definitions"; and
- 10 CFR 100.21, "Non-Seismic Siting Criteria," paragraphs (a), (b), and (h).

The applicable guidance for the evaluation of geography and demography is:

- DANU-ISG-2022-02, "Chapter 2, 'Site Information,'" March 2024 (ML23277A140);
- Regulatory Guide 4.7, Revision 4, "General Site Suitability Criteria for Nuclear Power Reactors." (ML23348A082).

2.1.1 Site

2.1.1.1 Technical Evaluation

The NRC staff evaluated the sufficiency of the PSAR information describing the geography of the site using the guidance and acceptance criteria from DANU-ISG-2022-02. The NRC staff referenced Google Maps and PSAR figures 2.1-3 and 2.1.2-1 to verify that the center point of the project area falls within the proposed site. The NRC staff also referenced Google Maps to independently verify the distances and directions specified in the PSAR to area boundaries, roads, railways, waterways, and other significant features surrounding the proposed site. A summary of the information provided in the PSAR that was verified by the NRC staff to determine the acceptability of the description of the site is below.

PSAR section 2.1.1.2 provides the coordinates of the center point of the project area. The site is nearly level to gently sloping coastal lowland, a typical landform developed from marine sediment deposits.

PSAR section 2.1.1.1 states that the proposed site for the LMGS is located in a coastal prairie of the Guadalupe River Delta in Calhoun County, Texas. The proposed facility and its supporting infrastructure will be located on an area of approximately 1,537 acres (6.2 square kilometers (km^2)) on land owned by The Dow Chemical Company (Dow) through its subsidiary Dow Hydrocarbons & Resources, LLC. The proposed site is approximately 22 miles (mi) (35 kilometers (km)) southeast of the City of Victoria, Texas and approximately 60 mi (97 km) north-northeast of the City of Corpus Christi, Texas. Eleven municipalities exist within 20 mi (32 km) of the proposed site. The City of Seadrift is the closest municipality. The City of Port Lavaca is the largest community near the proposed site. PSAR figure 2.1-2 shows the airports, roads, and railroads within 50 mi (80 km) of the proposed site. PSAR figure 2.1-3 shows that the proposed site can be accessed via Texas State Highways 185 (TX-185) and TX-35, and U.S. Route 87. A railyard and a spur connect the site to the Union Pacific Railroad, which connects Bloomington and the proposed site.

PSAR section 2.1.1 states that the proposed site is adjacent to Dow Chemical Seadrift Industrial railyard. As stated in PSAR section 2.2.2.6 and shown in PSAR figure 2.2-2, the proposed site is located approximately 2.3 mi (3.7 km) east of the Victoria Barge Canal. This canal connects the Port of Victoria with the Gulf Intra-coastal Waterway. Although the proposed facility will not use the canal for cooling the reactors, more than 400 barges use this canal annually to transport hazardous chemicals.

2.1.1.1 Conclusion

Based on the above, the NRC staff finds that PSAR section 2.1.1 and associated figures provide sufficient information to locate the proposed site. In addition, the NRC staff finds that the description of the proposed site is sufficiently detailed. In addition, the NRC staff finds that the description of the proposed site contains sufficient information to identify the types and locations of natural and human-made features and potential hazards on or near the site and the local, State, and Federal jurisdictions associated with the site and its surrounding area.

2.1.2 Exclusion Area Authority and Control

2.1.2.1 Technical Evaluation

The NRC staff evaluated the sufficiency of the PSAR information provided on authority and control of the exclusion area of the proposed facility. The NRC staff reviewed the PSAR to identify the exclusion area boundary (EAB) and to identify the operator of the LMGS and the entity with authority on lands within the EAB. The NRC staff further reviewed the PSAR to identify public roads, rail roads, and waterways traversing the EAB and commercial activities within the EAB. A summary of the information provided in the PSAR that was evaluated by the NRC staff to determine the acceptability of the authority and control of the EAB is below.

PSAR sections 2.1.1.1, 2.1.1.3, and 2.1.2 state that the EAB is 0.25 mi or 1,312 feet (ft) (400 meters (m)) from the radiological release envelope of the proposed facility. PSAR section 2.1.2 also states that the radiological release envelope is a rectangular area

circumscribing the reactor building (RB), fuel handling building, and helium service facility (HSF). The low population zone (LPZ) of the proposed facility coincides with the EAB. PSAR figures 1.1-3 and 2.1.2-1 show that there are no public roads, railroads, or waterways within the EAB. PSAR section 2.1.2 also states that access to the proposed site is controlled, and appropriate measures will be taken to prevent any pedestrian and vehicular intrusion into the EAB. PSAR section 2.1.2 further states that Long Mott Energy, LLC (LME) has selected a subsidiary of NextEra Energy Resources, LLC (NEER) to be the operator of LMGS and the operator will be responsible for controlling all activities within the EAB. Additionally, there are no residents within the EAB.

PSAR section 2.1.2.2 states that all recreational and hunting activities in addition to any agricultural use of the land within the owner-controlled area and the EAB are prohibited and will be controlled by the facility operator. PSAR section 2.1.2.2 also states that there are a few water bodies nearby, which are industrial basins, as shown in PSAR figure 1.1-3, within the EAB. Based on PSAR section 2.1.2.2, the NRC staff understands that Dow has total control over these industrial basins.

In summary, the NRC staff evaluated the sufficiency of the PSAR information provided on authority and control of the exclusion area of the proposed facility. The NRC staff finds that NEER, as the operator of the LMGS, will have absolute authority on all lands within the EAB from the pre-operational testing through plant operation. The NRC staff also finds that there are no public roads, rail roads, or waterways traversing the EAB and that Dow controls all commercial activities within the EAB and prohibits agricultural use, hunting, and recreational activities using Dow security personnel. Consequently, no arrangements for traffic control will be needed within the EAB as there are no public roads. The NRC staff also finds from PSAR figure 2.1.2-1 that there are no residences within the EAB and from PSAR figure 1.1-3 that there is no public access to nearby water bodies. Therefore, the NRC staff concludes that LME has identified the authority to administer control of all activities within the EAB.

2.1.2.2 Conclusion

Based on the above, the NRC staff finds that PSAR sections 2.1.1 and 2.1.2 provide sufficient information on the exclusion area authority and control for the proposed site. In addition, the NRC staff finds that the PSAR provides information that satisfies both acceptance criteria in DANU-ISG-2022-02, section 2.2.2.2, specifically a description of the legal authority for all activities within the designated exclusion area and a demonstration that the activities permitted within the exclusion area pose no hazard to the facility. The NRC staff therefore finds that the exclusion area control requirements or 10 CFR 100.21 and 10 CFR 100.3 are met.

2.1.3 Population Distribution

2.1.3.1 Technical Evaluation

The NRC staff evaluated the sufficiency of demographic information and acceptability of the selected site based on the existing and projected future population profile of the area surrounding the site using guidance in DANU-ISG-2022-02. The NRC staff reviewed the PSAR to identify the data sources used for population estimates and the methodology used to project population growth for the planned operational life of the facility. A summary of the information provided in the PSAR that was evaluated by the NRC staff to determine the acceptability of the estimated population distribution is below.

Resident Population

PSAR section 2.1.3 states that LME estimated the current population distribution surrounding the proposed LMGS site using the U.S. Census Bureau 2020 decennial census data. PSAR section 2.1.3.1.1 discusses the resident population within 5 mi (8 km), 10 mi (16 km), and 20 mi (32 km) from the proposed site. PSAR tables 2.1.3-3 and 2.1.3-4 provide the distribution of resident population up to 50 mi (80 km) from the proposed site. PSAR figures 2.1.3-2 through 2.1.3-12 plot the distribution of total population comprising both the resident and the transient population within 50 mi (80 km) of the proposed site. The NRC staff finds the use of the latest 2020 U.S. Census Bureau information on population is acceptable because it is an authoritative source.

PSAR section 2.1.3.4 identifies the City of Victoria as the nearest population center with more than 25,000 residents. PSAR table 2.1.3-8 shows the population of the City of Victoria was 65,534 in 2020 and is expected to grow steadily. The NRC staff confirmed that the City of Victoria is the nearest population center to the site. As this city has a population greater than 25,000 and is more than one and one-third distance times the distance from the reactor to the outer boundary of the LPZ, this city satisfies the definition of the population center in 10 CFR 100.3 and 100.11(a)(3).

Transient Population

The estimated transient population near the proposed site comes from several sources; for example, major employers, regional airports, schools, medical facilities, lodging facilities, and recreational areas. The applicant assumed that all users of these facilities would be transient visitors. The nearest transient population is at the L&W Frazier RV Park, approximately 0.6 mi (1.0 km) from the LPZ and emergency planning zone (EPZ) boundaries. PSAR section 2.1.3.2.1 discusses information on visitors developed from corporate and government websites, federal databases, tourism sites, and records of aggregators of service providers. PSAR table 2.1.3-5 lists the estimated transient population in 2020 within 20 mi (32 km) of the proposed site. Estimates of the raw transient population were weighted by the length of time each population group is expected to be in this area.

LME discussed in PSAR section 2.1.3.2.2 fifteen major recreational areas and 194 events and attractions with large venues within 20 to 50 mi (32 to 80 km) from the proposed site that generally draw large numbers of visitors to the region. PSAR table 2.1.3-6 lists the known facilities attracting large number visitors to this region. LME did not add this estimated transient population to the estimated resident population. Transient population was projected using the same approach as the resident population estimates for future years as discussed below.

The NRC staff finds that LME has used appropriate publicly available information to estimate the current and future transient populations associated with the regional airports, major employers, schools, medical facilities, lodging facilities, and recreational areas. The NRC staff finds that assuming that all users of certain facilities within 20 mi (32 km) of the proposed are transient populations is a conservative assumption as many users of these facilities may also be counted as residents. The NRC staff further finds that not adding visitors to major recreational areas and events between 20 and 50 mi (32 to 80 km) from the proposed site to the resident population is a reasonable assumption as this transient population is at a significant distance, more than 20 mi (32 km) from the proposed site, and these visitors would be in this region for a limited period of time.

Population Density

In PSAR section 2.1.3.5, LME describes the population density within 20 mi (32 km) surrounding the proposed facility. LME provided the estimated population density in PSAR table 2.1.3-7. PSAR figure 2.1.3-14 compares the projected population density, given in number of persons per square mile (ppsm), with the guidance in RG 4.7, "General Site Suitability Criteria for Nuclear Power Reactors," which recommends that the projected population density should not exceed 500 ppsm at the time of initial site approval and for 5 years thereafter. Based on PSAR figure 2.13-14, LME concluded that the projected population density is less than that specified in the regulatory guidance until the end of the operational life of the proposed facility. The nearest transient population is at the L&W Frazier RV Park, approximately 0.6 mi (1 km) from the LPZ and EPZ boundaries. Estimated raw transient populations were weighted based on the length of time each population group was expected to be within 20 mi (32 km) of the proposed site.

PSAR figure 2.1.13-14 plots the estimated population density in ppsm against a 20 mi (32 km) radial distance from the proposed site. The NRC staff finds that the estimated cumulative population density is less than 500 ppsm until 2070, which is consistent with the guidelines specified in RG 4.7. Therefore, the NRC staff concludes that the proposed reactor facility will be located away from densely populated centers.

Population Growth

Population distribution surrounding the proposed site was estimated for the years 2023, 2030, 2035, 2040, 2050, 2060, and 2070 using the county population projection dataset and the methodology from the Texas Demographic Center under the Institute of Demographic and Socioeconomic Research at the University of Texas at San Antonio. The annual population growth rates of the nearby counties were estimated using the information from the Census Bureau for years 2000 and 2020. Using this information, populations were projected for the 14 counties from a circle of 0.5 mi (0.8 km) to concentric circles up to 50 mi (80 km) from the site boundary. As shown in PSAR figures 2.1.3-1 and 2.1.3-2 and tables 2.1.3-1 and 2.1.3-2, population is estimated in concentric distance bands and in sixteen 22.5-degree wedge-shaped sectors.

The NRC staff finds that LME has used an acceptable methodology for the projected population for the planned operational life of the facility. The projected annual population growth rates are based on the 2000 and 2020 U.S. Census Bureau, and the population was projected county-wise using information and methodology from the Texas Demographic Center, which is an authoritative source.

Combined resident and transient population within 20 mi (32 km) of the proposed site is shown in PSAR figures 2.1.3-3 through 2.1.3-7 for years 2020, 2023, 2030, 2035, and 2070. The NRC staff finds that LME has used the latest 2020 U.S. Census data and projected the population data for future years using appropriate methods. Projected population information was presented for the year of facility approval and throughout the requested operating period of the proposed facility.

PSAR figure 2.1.3-13 shows the EAB and LPZ segments with a calculated population of 1 resident, although no resident or transient population exists within these areas. This is a result of the methodology used to project the population in each of the 22.5-degree segments. Based

on this information, the NRC staff concludes that LME can take protective measures for the population within the LPZ and also the EAB.

2.1.3.2 Conclusion

Based on the above, the NRC staff finds that PSAR section 2.1.3 provides sufficient information to describe the population distribution around the proposed site consistent with the guidance in DANU-ISG-2022-02. The NRC staff finds that the applicant has appropriately estimated both the resident and transient populations for the expected operating life of the facility. Additionally, the NRC staff find that the population density surrounding the proposed site is consistent with RG 4.7. Therefore, the NRC staff finds that the applicant provided sufficient information in the PSAR to meet 10 CFR 100.3 and 10 CFR 100.21(a), (b), and (h).

2.1.4 Conclusion

Based on its review, the NRC staff determined that the geographical and demographical information provided in the PSAR is sufficiently detailed and accurate to support assessments of the potential radiological impact on the public resulting from siting and operation of the proposed LMGS facility, including analyses (e.g., dose calculations) presented in other PSAR chapters. The NRC staff also finds that the geographic and demographic characteristics of the LMGS site do not render the site unsuitable for operation of the LMGS facility, as the site is located in an LPZ. The NRC staff also finds that the information provided meets the applicable acceptance criteria of DANU-ISG-2022-02, section 2.2.3.2. Accordingly, the NRC staff determined that the applicant meets the requirements in 10 CFR 50.34(a)(1)(ii), 10 CFR 100.3, and 10 CFR 100.21.

2.2 Nearby Industrial, Transportation, and Military Facilities

This section of the SE identifies, locates, and describes present and projected facilities and transportation routes in the vicinity of the LMGS site. The facilities and transportation routes are then evaluated to determine the effects of potential accidents in the vicinity of the LMGS site and whether hazards due to the facilities and transportation routes need to be considered design-basis events for plant design. Parameters used as the basis for the evaluations are provided below.

LME listed the industrial facilities within 5 mi (8 km) of the proposed site in PSAR section 2.2.1. In addition, LME described in PSAR section 2.2.2 the potential hazardous materials that are either manufactured or stored at these facilities or transported using nearby transportation routes. PSAR section 2.2.3 describes the analyses conducted and results obtained on whether any of these external human-induced hazards can become a design-basis hazard to the proposed facility. The NRC staff reviewed the description of the facilities and the analyses conducted to determine whether LME used appropriate methods and analyses to screen out each potential hazard. LME analyzed the following accident categories associated with nearby facilities and transportation routes to identify any design-basis hazards: (1) aircraft crashes, (2) explosion (detonation) from immediate ignition, (3) explosion (deflagration) from delayed ignition of vapor cloud, (4) fires, and (5) toxic chemicals.

The applicable regulatory requirements related to nearby industrial, transportation, and military facilities are as follows:

- 10 CFR 50.34(a)(1)(ii);
- 10 CFR 100.20(b); and
- 10 CFR 100.21(e).

The applicable guidance for the evaluation of nearby industrial, transportation, and military facilities is:

- DANU-ISG-2022-02, “Chapter 2, 'Site Information,'” March 2024 (ML23277A140);
- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Plants: [Light-Water Reactor] LWR Edition.” (ML070810350);
- RG 1.91, Revision 3, “Evaluations of Explosions Postulated to Occur at Nearby Facilities and on Transportation Routes near Nuclear Power Plants.” (ML21260A242);
- NUREG–0570, “Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release.” (ML063480551);
- RG 1.145, “Auditing of Quality Assurance Programs for Nuclear Power Plants.” (ML003740205);
- RG 1.78, Revision 2, “Evaluating the Habitability of a Nuclear Power Plant control Room During a Postulated Hazardous Chemical Release.” (ML21253A071);
- RG 1.233, Revision 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 NEI 18-04, Revision 1, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” (ML19241A472), with clarifications; and
- RG 1.253, Revision 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 NEI 21-07, Revision 1, “Technology Inclusive Guidance for Non-Light Water Reactors, Safety Analysis Report Content: For Applicants Using the NEI 18-04 Methodology,” (ML22060A190), with clarifications and additions.

2.2.1 Aircraft Crash Hazard

The NRC staff reviewed, following section 3.5.1.6 of NUREG-0800, LME’s assessment of potential hazards from aircraft flying nearby to assess whether aircraft crashes onto the proposed facility can be identified as a design-basis hazard.

2.2.1.1 Technical Evaluation

LME described the potential aircraft crash hazard at the proposed facility in PSAR section 2.2.2.7. The NRC staff reviewed the discussion provided in PSAR section 2.2.2.7 using guidance in section 3.5.1.6 of NUREG–0800 (NRC, 2007). Additionally, the NRC staff used the guidance given in the Department of Energy (DOE) Standard DOE–STD–3014–2006 (DOE, 2006) in this review. The methodology in the DOE standard is consistent with NUREG-0800 and provides updated information to perform aircraft crash analysis. The NRC staff also audited LME's calculation document "Aircraft Hazard Evaluation," XE-C-ME-022, Revision 1, 2024, by Sargent & Lundy to confirm that the calculations support the information presented in the PSAR.

Nearby Airports

LME listed nearby airports to the proposed site in PSAR table 2.2-8. This table also provides the distance to each airport from the proposed site and annual number of operations (take-offs and landings). PSAR figure 2.2-5 shows locations of these airports with respect to the proposed site. Based on information in PSAR table 2.2-8, only Whatley Flying Service (8TA1) is within 5 mi (8 km) of the proposed site and only aircraft used for occasional crop dusting (i.e., less than 25 times in a year) use this airfield.

In PSAR section 2.2.2.7, LME identified two private airports within 10 mi (16 km) of the proposed site: Green Lake Ranch (69TX) and Tanners (9TE4). PSAR table 2.2-8 shows no operations taking place at the Green Lake Ranch airport. Only general aviation aircraft use Tanners airport with 2,200 operations annually. Calhoun County Airport (PKV) is the only public airport within 10 mi (16 km) from the proposed site. Other public airports, such as Sterling airport (10XA), Victoria Regional Airport (KVCT), William P. Hobby Airport (HOU), and George Bush Intercontinental Airport (IAH) are beyond 10 mi (16 km) from the proposed site.

LME used the Federal Aviation Administration (FAA) operational information to determine the annual number of operations at these airports. LME screened out all airports, except Watley Flying Service based on the number of annual operations and proximity to the proposed site. The NRC staff accepts LME's screening of airports based on information in PSAR table 2.2-8, which indicates that all airports listed have annual flight operations lower than that needed to pose a credible hazard to the proposed facility based on the proximity criterion 1.A of section 3.5.1.6 of NUREG–0800. In addition, the NRC staff concludes that because of the limited number of annual operations of the crop duster aircraft using the Whatley Flying Service airport as well as their flight elevation close to the ground, these aircraft do not pose a credible hazard to the proposed facility.

Nearby Airways

PSAR figure 2.2-5 shows the Federal airways near the proposed site. In PSAR section 2.2.2.7.2, LME discussed hazards to the proposed facility from aircraft crashes while transiting nearby Federal airways. LME listed two low-altitude Victor Airways, V13–V407 and V20–V70, and one Jet Route J22–J29, near the proposed site. The centerline of the Victor Airway V13–V407 is approximately 1 mi (1.6 km) from the proposed site boundary; the centerline of the Victor Airway V20–V70 is approximately 7.3 mi (12 km) from the proposed site boundary; and the centerline of the Jet Route J22–J29 is approximately 4.6 mi (7.4 km) from the proposed site boundary. In addition, the military airway VR–151 is more than 10 mi (16 km) away from the proposed site. LME concluded that the airways V13–V407 and J22–J29 do not

satisfy the proximity criterion 1.C of section 3.5.1.6 of NUREG–0800. Consequently, and consistent with the guidance in NUREG-0800, LME conducted a detailed analysis of aircraft crash hazard to the proposed facility of airways V13–V407 and J22–J29, and the NRC staff's evaluation of that analysis is below.

Annual Crash Frequency

LME used DOE-STD-3014-2006 (DOE, 2006) to estimate the annual crash frequency from aircraft. The NRC staff finds that the method presented in the DOE standard is acceptable because it is consistent with section 3.5.1.6 of NUREG-0800. The NRC staff notes that the aircraft crash frequency should be below 1.0×10^{-7} per year to be considered as a beyond-design-basis hazard, consistent with section 2.2.3 of NUREG-0800.

As the proposed site is not close to any of the DOE facilities listed in DOE (2006), LME used the parameter values of "Average CONUS [continental United States]" to estimate the annual crash frequency from flights by all types of aircraft (i.e., general aviation, air carrier, air taxi, large military, and small military) transiting these airways. LME estimated the annual crash frequency to the proposed facility to be 2.97×10^{-6} crashes from all types of aircraft flying these airways, as stated in PSAR section 2.2.2.7.2. Additionally, a major portion of the total aircraft crash hazard, 2.85×10^{-6} crashes per year, is from general aviation aircraft on these airways, as discussed in PSAR section 2.2.2.7. Therefore, based on the above discussion the NRC staff finds that aircraft flying in the vicinity would pose a credible hazard to the proposed facility because the estimated crash frequency is above 1.0×10^{-7} crashes per year, and would need to be accounted for in designing the safety-related (SR) and certain non-safety-related with special treatment (NSRST) structures, systems and components (SSC) at the proposed facility.

LME recognizes this in PSAR section 2.2.2.7.2 and commits to performing a risk-informed assessment of accidental aircraft impact hazards in accordance with the non-light-water reactor (non-LWR) probabilistic risk assessment (PRA) Standard (ASME, 2021) and NEI 18-04 (NEI, 2019) and providing it with the operating license (OL) application. LME did not address aircraft impact as a design-basis hazard in PSAR section 6.1.1 at the construction permit (CP) stage because of their expectation that it would be screened out with the more detailed analysis at the OL application. Based on the level of conservatism in the data used in the analysis, the NRC staff agrees and determined that this is consistent with the guidance in NEI 21-07, which indicates that the design-basis hazard information provided in the PSAR should be sufficient to support demonstrating the viability of the design. RG 1.253, Appendix A also indicates that external hazards PRAs are not required to meet the requirements of a minimally acceptable PRA at the CP stage.

If the risk-informed assessment of accidental aircraft impact hazards performed for the OL application shows aircraft impact does not screen out and is a design-basis event, it will need to be addressed in the final design and safety analysis to show that safety-significant SSCs are designed to withstand the event or that the dose consequence is acceptable. The NRC staff determined that this may also be a viable path based on the robustness of the reactor shield structure and spent fuel intermediate storage structure.

2.2.1.2 Conclusions

The NRC staff finds that the estimated annual crash hazard has several conservative assumptions. For example, the crash rate for each aircraft type comes from the DOE standard. The aircraft crash rates in this standard were derived from actual aircraft crashes that occurred before 1998. Advanced design of current aircraft with better training and other improvements in the aviation industry make the current fleet of aircraft less crash prone, reducing the crash rate (number of crashes per mile of flight). In addition, the crash rates used were derived from crashes that did not take place near the proposed facility. All non-airport crashes in the lower 48 states were used to derive the crash rates in the DOE standard. Therefore, the crash rate includes crashes from areas that are crash prone, such as areas with much higher flight density than South Texas.

Based on the level of conservatism in the current screening analysis and the robustness in the preliminary design, the NRC staff determined that sufficient information was provided in the PSAR to meet 10 CFR 50.34(a)(1)(ii), 10 CFR 100.20(b), and 10 CFR 100.21(e), and that the applicant will be able to demonstrate viable paths for addressing aircraft impact at the OL stage. The final disposition of the aircraft crash hazard would occur at the OL stage when LME would need to show, using risk-informed assessment or through robustness of the facility design, that accidental aircraft crashes do not pose a credible hazard to the proposed facility.

2.2.2 Description of Nearby Facilities and Transportation Routes

2.2.2.1 Technical Evaluation

The NRC staff reviewed the application to determine whether LME sufficiently evaluated potential hazards associated with nearby transportation routes, industrial and military facilities, and civilian and military airports to meet 10 CFR 100.20(b) and 10 CFR 100.21(e). PSAR tables 2.2-3 through 2.2-7 list the explosive and toxic chemicals that are transported by trucks, pipelines, railcars, and barges near the proposed site. The NRC staff reviewed the PSAR to determine whether the descriptions and characteristics of nearby facilities, highways, railroads, pipelines, and waterways were included in the application. A summary of the information provided in the PSAR that allowed the NRC staff to conclude that the application gives a complete and current overview of the facilities, activities, and materials located in or transported through the vicinity of the proposed site, as discussed in DANU-ISG-2022-02, is below.

Nearby Facilities

PSAR table 2.2-1 lists major projects of the Dow Seadrift Operations (SDO), Braskem, Calhoun Chemical, and INEOS Nitriles chemical plants located near the proposed site. Dow SDO is adjacent (within 0.5 mi) to the proposed site and is located just south of Texas highway TX-35 along TX-185, as stated in PSAR section 2.2.2.2.1. Dow SDO manufactures polyethylene, glycols, and oxide derivatives. PSAR table 2.2-2 lists the chemicals stored or manufactured at this facility along with their lower explosive limit (LEL) and toxicity limit. Dow SDO uses a barge dock on the Victoria Barge Canal and two large, connected rail yards of the Union Pacific (UP) and the Burlington Northern Santa Fe (BNSF). In addition, this facility uses more than ten (10) dedicated pipelines for transporting chemicals.

Braskem USA operates one of the plants at the Dow SDO facility and is approximately 1.2 mi (1.9 km) from the proposed site, as stated in PSAR section 2.2.2.2.2. As Braskem USA

produces polypropylene in one of the plants of the Dow SDO facility, hazards from toxic and explosive chemicals at this facility are bound by the chemicals listed under Dow SDO.

Seadrift Coke LP produces premium needle coke. PSAR table 2.2-2 lists petroleum naphtha as the toxic and explosive chemical at Seadrift Coke LP facility. Calhoun Chemical LLC is approximately 3.6 mi (5.8 km) northwest of the proposed site along highway TX-185. It manufactures polyacrylamide, as stated in PSAR section 2.2.2.2.4. PSAR table 2.2-2 lists the toxic and explosive chemicals produced and stored at this facility. INEOS Nitriles is located approximately 4.8 mi (7.7 km) northwest of the proposed site along TX-185 and manufactures acrylonitrile, as stated in PSAR section 2.2.2.2.5.
Frontier Logistics is a trucking company located approximately 2.1 mi (3.4 km) from the proposed site. As stated in PSAR section 2.2.2.2.6, Frontier Logistics ships only the non-toxic and non-explosive end products from Dow SDO. Products from the other facilities in the area are transported by Kenan Advantage Group, located approximately 2.7 mi (4.3 km) from the proposed site.

In summary, the NRC staff finds the applicant's assessment of whether bulk storage of hazardous materials at facilities near the proposed facility site could pose potential explosion hazardous is acceptable because LME followed RG 1.91 to identify the chemicals that have potential to explode. The applicant collected information on quantity of chemicals stored at these facilities. In addition, the NRC staff reviewed the above information and determined that the facilities located in the vicinity of the proposed site are adequately described in the PSAR such that nature of any hazards posed to the facility can be assessed.

Highways

Two highways, TX-35 and TX-185, are within 5 mi (8 km) from the proposed site. TX-35 connects Houston to Corpus Christi and passes through Port Lavaca and Point Comfort. At its closest point, TX-35 is approximately 0.9 mi (1.4 km) from the proposed site. TX-185 connects Port O'Connor with Victoria and is approximately 1.3 mi (2.1 km) away from the proposed site at its closest point, as stated in PSAR section 2.2.2.4.

LME contacted the chemical facilities along TX-35 and TX-185 for information on hazardous cargoes and also reviewed their Environmental Protection Agency (EPA) Tier II reports. PSAR table 2.2-5 lists the explosive and toxic hazardous chemicals most commonly transported on TX-35 and TX-135. Following the guidance in RG 1.91, LME listed in PSAR table 2.2-5 the maximum quantity of any chemicals transported on highway trucks to be 50,000 pounds (lb) (22,680 kilograms (kg)), unless maximum quantities of the chemical being transported are known specifically. Chemicals transported to and from Dow SDO and other nearby sites were excluded from PSAR table 2.2-5 as the quantity of these chemicals stored on-site would be bounding.

In summary, the NRC staff finds the applicant's input to determining whether bulk storage or transportation of hazardous materials may occur at or near the site and its assessment of the impact of potential explosions based on hazardous cargoes transported on highways near the proposed site acceptable because LME followed RG 1.91 to identify the quantity of chemicals transported on highway trucks. In addition, the NRC staff reviewed the above information and determined that the highways located in the vicinity of the proposed site are adequately described in the PSAR such that nature of any hazards posed to the facility can be determined.

Railroads

The DOW SDO facility has a large railyard, approximately 0.3 mi (0.5 km) from the proposed site. This railyard is connected to both the BNSF and the UP tracks. At their closest points, both tracks are approximately 1.4 mi (2.3 km) from the proposed site. The BNSF track runs from Karney, Texas, to slightly south of TX-35, and connects to the north railyard. The UP track connects Victoria, Texas and Dow SDO.

As stated in PSAR section 2.2.2.5, approximately 4,800 hazardous cargo shipments are transported annually by both the UP and BNSF rail lines. LME listed in PSAR table 2.2-6 the most frequently transported hazardous explosive and toxic materials by rail. Following the guidance in RG 1.78, chemicals shipped less than 30 times annually are not listed in PSAR table 2.2-6, as these are considered not frequent.

In summary, the NRC staff finds that the applicant's assessment of potential hazards from bulk transportation of hazardous materials near the proposed site is acceptable because LME followed RG 1.91 to identify the quantity of materials transported on rail cars. In addition, the NRC staff determined that the rail lines are adequately described in the PSAR such that nature of any hazards posed to the facility can be determined.

Pipelines

As stated in PSAR section 2.2.2.3, more than 25 pipelines carrying natural gas, methane, liquefied petroleum gas (LPG), anhydrous ammonia, propylene, and acrylonitrile are located within 5 mi (8 km) of the proposed site. These pipelines include both transmission and gathering natural gas pipelines.

PSAR table 2.2-3 lists the gas pipelines operated by Seadrift Pipeline Corporation of Dow, Enterprise Products Operating LLC, Kinder Morgan Tejas Pipeline, and Enerfin Field Services, LLC. PSAR table 2.2-4 lists pipelines carrying liquid products near the proposed site. Seadrift Pipeline Corporation, Enterprise Products Operating LLC, and INEOS USA LLC operate the liquid products pipelines located within 5 mi (8 km) of the proposed site.

PSAR table 2.2-3 lists both methane and ethylene gas pipelines near the proposed site. As natural gas is mostly methane, methane and natural gas are interchangeably used. A 30-inch (in.) (76-centimeter (cm)) nominal diameter natural gas pipeline operated by Enterprise Products Operating LLC is at a distance of approximately 0.59 mi (0.95 km) from the proposed site. It is the closest and also the largest diameter natural gas pipeline within 5 mi (8 km) of the proposed site. It operates at a maximum pressure of 936 pounds per square inch gauge (psig) (6.45 megapascals (MPa)), as detailed in PSAR table 2.2-3. A 12-in (30-cm). diameter ethylene pipeline, operated by Seadrift Pipeline Corporation, is located approximately 0.5 mi (0.8 km) from the proposed site.

As listed in PSAR table 2.2-4, the largest LPG pipeline operated by Seadrift Pipeline Corporation near the proposed site has a diameter of 12 in. (30 cm) with a maximum operating pressure of 990 psig (6.83 MPa) at approximately 1 mi (1.6 km) from the proposed site. The nearest LPG pipeline to the proposed site is the 6-in. (15-cm) diameter pipeline of Seadrift Pipeline Corporation at approximately 0.43 mi (0.69 km) from the proposed site. The LPG pipeline of Enterprise Products Operating LLC is 8 in. (20 cm) in diameter and approximately

0.6 mi (1.0 km) from the proposed site, as listed in PSAR table 2.2-4. Other LPG pipelines have smaller diameters and operate at higher pressures of approximately 2,160 psig (14.9 MPa).

The propylene pipeline operated by Seadrift Pipeline Corporation is 0.9 mi (1.4 km) away and has a diameter of 3 in. (7.5 cm). The nearest Enterprise Products propylene pipeline is 0.59 mi (0.95 km) away with a diameter of 6 in. (15 cm) INEOS pipelines carrying liquid (anhydrous) ammonia and acrylonitrile are 8 in. (20 cm) in diameter and are located approximately 3.6 mi (5.8 km) away from the proposed site.

In summary, the NRC staff finds that the applicant has identified and provided appropriate information on the pipelines carrying hazardous chemicals near the proposed site. This information can be used to assess potential hazards to the proposed facility from a release of hazardous chemicals.

Waterways

The proposed facility is located approximately 2.3 mi (3.7 km) east of the Victoria Barge Canal. This canal is approximately 35 mi (56 km) long and connects the Gulf Intra-Coastal Waterway with the Port of Victoria. It is 125 ft (38 m) wide and at least 12 ft (3.7 m) deep. As stated in PSAR section 2.2.2.6, the proposed facility will not use any water from this waterway for cooling. Consequently, collision of barge traffic with the intake structure is not a credible hazard to the proposed facility. However, barge traffic transporting hazardous cargo can potentially pose a hazard to the proposed facility and is reviewed in this section.

As stated in PSAR section 2.2.2.6, LME obtained the information regarding the hazardous commodities transported along the canal from the Port of Victoria and industrial facilities having docks on the canal. More than 400 barges with hazardous chemicals are transported annually on this canal, with each barge generally containing more than 3,000,000 lb (1,360,000 kg) of chemicals. The toxic and explosive chemicals most frequently transported via this canal are listed in PSAR table 2.2-7. However, chemicals shipped fewer than 50 times per year are not presented in this table, consistent with the definition of “frequent” detailed in the guidance in RG 1.78.

In summary, the NRC staff finds the applicant’s approach to identifying frequent transport of toxic and explosive chemicals acceptable, as it is consistent with the guidance in RG 1.78. In addition, the NRC staff reviewed the above information and determined that the waterways located in the vicinity of the proposed site are adequately described in the PSAR such that that nature of any hazards posed to the facility can be determined.

Nearby Industrial Growth

LME consulted the Victoria Economic Development Corporation (VEDC) to gain insights into the projected future growth of industrial activities near the proposed site. VEDC responded to industrial and manufacturing inquiries for Calhoun, Jackson, Goliad, and Victoria counties and the NRC staff concludes that VEDC is an authoritative source for information on future industrial growth near the proposed site because VEDC works directly with existing and future businesses in the region and is therefore a reliable and knowledgeable source of information on future growth of industrial activities near the proposed site.

PSAR section 2.2.2.9 states that there are plans to widen and deepen the Matagorda and Lavaca Bay shipping canal to bring in larger ships to dock at the Seahawk Oil Terminal in Point Comfort; however, this canal is more than 5 mi (8 km) from the proposed site and the potential to generate an overpressure or a thermal hazard is negligible consistent with NUREG-0800.

As stated in PSAR section 2.2.2.9, Lynas Rare Earths will construct a rare earth processing facility at Seadrift, Texas. The NRC staff notes that Lynas Rare Earths has purchased a 49-acre greenfield site from Dow SDO (Lynas Rare Earths, 2026). This plot of land is approximately 1 mi (1.6 km) northeast of the proposed site. The plan is to develop a circular-shaped mine to extract and process rare earth minerals to supply magnets.

Both First Ammonia and ACME Ammonia are planning ammonia production facilities more than 5 mi (8 km) away from the proposed site. In addition, ACME is considering a hydrogen production facility at the Port of Victoria Industrial Park, more than 5 mi (8 km) from the proposed site along the Victoria Barge Canal. Although hydrogen is a flammable gas and can produce significant overpressure, a distance more than 5 mi (8 km) from the proposed site is sufficient to dilute the gas below its LEL and not produce an overpressure event that could impact the proposed site. Similarly, ammonia is a toxic substance that can affect personnel outside or inside an enclosed building, such as the control room of the proposed facility, without sufficient protection through filtration of the heating, ventilation and air conditioning (HVAC) intake. However, a distance of 5 mi (8 km) from the proposed site is sufficient to dilute the ammonia concentration below its immediately dangerous to life or health value of 300 ppm, following RG 1.78. In addition, as stated in PSAR section 2.2.3.5, the proposed facility does not require operator actions from the control room. Therefore, the NRC staff finds that ammonia production plants at the planned distances from the proposed site would not produce a credible hazard to the proposed facility.

As stated in PSAR section 2.2.2.9, the Alcoa Superfund site in Point Comfort is being remediate and will likely be used in the future for industrial ventures. Although the site is greater than 5 mi (8 km) from the proposed site, it is along highway TX-35, and transportation of hazardous materials along this route could impact the proposed site.

2.2.2.2 Conclusion

The NRC staff has reviewed the information presented in PSAR section 2.2.2.2 regarding how potentially hazardous materials stored or manufactured in facilities and transported along transportation routes, such as roadways, railways, and waterways near the proposed site, that may pose a credible hazard to the proposed facility are designed to mitigate the hazard. The NRC staff finds that LME has used appropriate guidance to identify potential hazardous materials and described the activities requiring assessment of whether they pose a credible hazard to the proposed facility.

The NRC staff finds that LME has appropriately identified and assessed the potential hazards associated with nearby transportation routes, industrial and military facilities, and civilian and military airports and determined that the risk is very low and, therefore, has satisfied the requirements of 10 CFR 100.20(b) with respect to those hazards. In addition, the NRC staff finds that LME evaluated the potential hazards associated with nearby transportation routes, industrial and military facilities, and civilian and military airports associated with nearby facilities and transportation routes and showed that the hazards do not pose undue risk to the proposed facility and, therefore, has satisfied the requirements of 10 CFR 100.21(e) with respect to those

hazards. Additionally, as LME has conducted a safety assessment of the proposed site and facility and showed that the potential for accidents that could result in the release of significant quantities of radioactive fission products is extremely low with respect to potential hazards associated with nearby transportation routes, industrial and military facilities, and civilian and military airports, the NRC staff finds that the requirements of 10 CFR 50.34(a)(1)(ii) have been satisfied with respect to those hazards.

2.2.3 Explosion (Detonation) from Immediate Ignition

The NRC staff reviewed the information presented in PSAR sections 2.2.1, 2.2.2, and 2.2.3 using NUREG–0800 sections 2.2.1 - 2.2.2 and 2.2.3 to evaluate LME's assessment of hazards associated with the storage or operation of chemicals at nearby facilities and the movement of these chemicals using the transportation routes to establish whether the plant design can accommodate these hazards. LME generally used deterministic methods to assess these hazards; however, rate of occurrence of potential exposure is also estimated in selected cases to demonstrate that the postulated type of accident is extremely rare (e.g., below the regulatory threshold detailed in section 2.2.3 of NUREG–0800). However, any event that cannot be shown to be rare following section 2.2.3 of NUREG–0800 will be a design-basis event.

The NRC staff reviewed the discussion in PSAR section 2.2.3 and audited calculation packages with details on the methodologies used, scenarios considered, assumptions made about the scenarios, and results of hazard analyses.

2.2.3.1 Technical Evaluation

Immediate ignition of released flammable liquids and gaseous fuels can result in an explosion generating overpressure (blast pressure), which is a concern for structural response at the proposed facility. LME evaluated the blast overpressure generated from explosion of chemicals (sources) near the proposed facility that would have an adverse effect on plant operation or would prevent safe shutdown of the reactors. PSAR section 2.2.3.2 summarizes the methods used to assess the impact of explosions on the facility and results of the assessment.

LME assessed the hazards from materials stored or manufactured at the nearby facilities or transported on nearby transportation routes to screen in those that need to be treated as a design-basis hazard. PSAR section 2.2.3 describes LME's analysis of ten explosions selected to bound the explosive chemicals located near the proposed site. LME stated that the explosions evaluated meet the acceptance criteria in RG 1.91 and are not identified as design-basis hazards.

The NRC staff reviewed LME's assessment against the guidance in RG 1.91. LME addressed the explosion hazards using the trinitrotoluene (TNT) equivalency of the selected chemical using its heat of combustion value, listed in Factory Mutual 7-42 (2008). LME used the TNT equivalency method, as described in RG 1.91, to calculate the blast overpressure. The mass of a material that would produce the same blast effects as a mass of TNT was estimated. LME used the guidance in RG 1.91 to select the overpressure level of 1 pounds per square inch (psi) (6.9 kilopascal (kPa)) to determine whether the generated overpressure would be damaging to the SR structures at the proposed facility.

LME used a "confined" blast scenario to represent an explosion of flammable gas in a tank at ambient pressure at its upper explosive limit (UEL). LME assumed that the entire content of the

tank would participate in the explosion and the explosive yield fraction in equation (3) of RG 1.91 would be 100%. In addition, chemicals, which are gases in atmospheric conditions, can be released entirely to the surrounding atmosphere through a leak in the storage tank or after the tank fails catastrophically. LME used the term “semi-confined” scenario instead of “unconfined” scenario for explosions of materials released from a tank as flammable vapors, where only a fraction of the released chemical actually takes part in the explosion process. Recommended values of the explosion yield factor, as given in table 1 of RG 1.91, were used based on whether the explosion takes place in a confined or unconfined configuration.

In some instances, these deterministic approaches with the worst-case conditions show a potential to generate an overpressure exceeding 1 pound per square inch differential (psid) at the nuclear island (NI) area of the proposed facility. When the hazard did not screen out using the deterministic approach, LME assessed the annual frequency of occurrence of overpressure exceeding 1 psid (6.9 kPa) and showed it to be a rare event (less than 1×10^{-6} occurrence annually) so that these events do not pose a credible hazard to the proposed facility. LME assessed whether a particular event at a facility would generate an overpressure damaging to SR structures at the proposed facility in two ways. Following RG 1.91, if the estimated overpressure of 1 psid (6.9 kPa) occurs at a distance smaller than that between the source location (location of availability or release) and NI area of the proposed facility, the structures would be safe. Alternatively, if the concentration of a chemical estimated at the NI location of the proposed facility is less than its LEL value, there will not be an explosion overpressure generated. For some of the events analyzed, LME used a probabilistic approach to demonstrate that the event is rare, with a frequency of occurrence below 1×10^{-6} annually.

The hazard resulting from confined and semi-confined explosion event was assessed by determining the distance to the location where the resulting overpressure would be 1 psid (6.9 kPa) or the concentration of the chemical in the atmosphere would be just at the LEL of the specific chemical. If the actual distance between the safety-significant structures at the proposed facility and the release location is more than the calculated distance, then the release of the specific chemical would not be a credible hazard to the proposed facility.

As described in PSAR section 2.2.3.2, LME evaluated ten bounding explosion scenarios to analyze all potential explosion hazards at or near the proposed site. These bounding scenarios are summarized below:

Case 1: A tank at the Dow SDO releases 50 lb (23 kg) of hydrogen. Immediately upon release, the hydrogen detonates after encountering an ignition source. As stated in PSAR section 2.2.3.2, the standoff distance to the 1-psi (6.9 kpa) limit is estimated to be less than 500 ft (152 m), which is significantly smaller than 3,000 ft (914 m), the distance between the proposed LMGS NI and the hydrogen tank at Dow SDO. This case represents the worst-case scenario of hydrogen explosion (detonation) nearby affecting the proposed LMGS facility and the NRC staff finds that this potential event would not be a design-basis hazard.

Case 2: A tank releases 50,000 lb (22,680 kg) of hydroxy carbon. After release from the tank, the hydrocarbon plume meets an ignition source and detonates producing an overpressure. A value of 15% is assumed to be the yield factor, the maximum value as given in table 1 of RG 1.91 for unconfined vapor clouds. The standoff distance to the 1-psi (6.9 kPa) overpressure was estimated to be 1,970 ft (600 m) and bounds all transport of hazardous chemicals by

highway trucks. Therefore, potential detonation of hydroxy carbon while being transported on the nearby roads would not be design-basis hazard to the proposed facility.

Case 3: A tank instantly releases 294,000 lb (133,000 kg) of hydroxy carbon, which is the capacity of the largest storage tank at Dow SDO, and the hydroxy carbon detonates after encountering an ignition source. Assuming a yield factor of 15%, the standoff distance to the 1 psid (6.9 kPa) overpressure was estimated to be 3,560 ft (1,085 m) using the TNT equivalency method, which is less than 3,800 ft (1,158 m), the separation distance between this tank and proposed LMGS NI. Therefore, the NRC staff finds that a hypothetical release of the entire content of the largest hydroxy carbon tank at Dow SDO would not produce a credible hazard at the proposed site and bounds all offsite tanks at nearby facilities except those evaluated in Case 4.

Case 4: A tank of 5,000,000 lb (2,268,000 kg) of hydroxy carbon evaluated as a semi-confined explosion bounds the inventory at the nearby facilities outside of Dow SDO, such as INEOS and Seadrift Coke, and the hazardous chemicals transported by barges on the Victoria Barge Canal. Assuming the entire mass of hydroxy carbon detonates immediately upon release, using the TNT Equivalency method, LME estimated the standoff distance to the 1 psid (6.9 kPa) overpressure would be 9,150 ft (2,789 m). This distance is less than the distance from the large tanks at INEOS and Seadrift Coke facilities to the proposed NI location and much less than the distance from the Victoria Barge Canal to the proposed NI location. Therefore, the NRC staff finds that potential detonation of hydroxy carbon at other nearby facilities except Dow SDO would not be a design-basis hazard to the proposed facility.

Case 5: A tank containing 294,000 lb (133,000 kg) of hydroxy carbon at Dow SDO detonates resulting in a confined explosion with the yield factor equal to 100%. Using the TNT equivalency method, the standoff distance to the 1-psi (6.9 kPa) overpressure limit was estimated to be 1,690 ft (515 m). This standoff distance is less than the actual distance (more than 3,000 ft (914 m)) between the LMGS and Dow SDO. PSAR section 2.2.3.2 states that Case 5 bounds the largest tank at Dow SDO, which has a capacity of 5,400,000 lb (2,450,000 kg) of ethylene glycol monoethyl ether based on specific density of the chemical and its UEL. Therefore, the NRC staff finds that detonation of entire inventory of hydroxy carbon at Dow SDO would not be a design-basis hazard at the proposed facility.

Case 6: A tank containing 13,500,000 lb (22,680 kg) flammable gas detonates resulting in a confined explosion with the yield factor equal to 100%. The standoff distance to the 1-psid (6.9 kPa) overpressure was estimated to be 6,060 ft (1,847 m). This tank represents the largest tank outside of Dow SDO at INEOS. As stated in Case 4, this standoff distance is less than the actual distances between the LMGS and Dow SDO, INEOS, and Seadrift Coke. Therefore, the NRC staff finds that detonation hazard from all hydrocarbon tanks at these facilities would be bounded by this case.

Case 7: A hypothetical tank containing 13,500,000 lb (6,120,000 kg) of propylene undergoes a boiling liquid expanding vapor explosion (BLEVE) event. The standoff distance to the 1-psid (6.9 kPa) overpressure limit is estimated to be 2,350 ft (716 m), which is less than the distance of the proposed LMGS from Dow SDO chemical sources. Additionally, tanks at other nearby facilities are further away. Therefore, the NRC staff finds that potential BLEVE events at any of the nearby facilities would not be a credible hazard to the proposed facility.

Case 8: LME estimated the standoff distance to the 1-psid (6.9 kPa) limit from TNT stored at Dow SDO to be 785 ft (239 m), which is less than the distance from the TNT to the LMGS NI. Consequently, the NRC staff finds that TNT stored at Dow SDO would not produce a credible hazard to the proposed facility.

Case 9: A truck carrying 50,000 lb (22,680 kg) of TNT explodes on a nearby local road. The standoff distance to the 1-psid (6.9 kPa) overpressure is estimated to be 1,660 ft (506 m), which is less than 4,000 ft (1,219 m), the distance of the LMGS from the nearest road. Per RG 1.91, 50,000 lb (22,680 kg) of chemicals is the maximum quantity that can be transported by trucks on U.S. roads to replenish the stock at Dow SDO. Therefore, the NRC staff finds that transportation of solid explosives, such as TNT, on roads near the proposed site does not pose a hazard to the proposed facility.

Case 10: Railcars in the railyard of Dow SDO carry solids that may become aerosolized as dust. This dust cloud can explode. Assuming that 10% of the maximum allowable cargo would be aerosolized as dust, the standoff distance to the 1-psid (6.9 kPa) overpressure limit is estimated to be 1,290 ft (393 m) using the TNT equivalency method. This standoff distance is less than 1,500 ft (457 m), the distance of the LMGS NI to the closest railyard. The NRC staff notes that this fraction depends entirely on the material's state and the forces acting on it. DOE Handbook, Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities (DOE-HDBK-3010-94; DOE, 1994) uses a standardized bounding estimate of less than 0.5% from a free fall slip or drop of fine solids. Based on this, the NRC staff finds that the assumption that 10% of the cargo aerosolized and participated in a detonation event is conservative as the expected portion of mass becoming aerosolized in the rail yard is much less.

2.2.3.2 Conclusion

The NRC staff finds it acceptable that LME assumed the maximum yield factor of 15% in selecting the bounding scenarios because the expected yield factor (e.g., those listed in table 1 of FM Global (2008)) of most chemicals is significantly less. Additionally, the size of the full hydroxy carbon tank outside of Dow SDO is extremely large at 5,000,000 lb (2,268,000 kg), which also bounds hazardous chemicals transported by barges in the Victoria Barge Canal. However, as Dow SDO is close to the proposed facility, a separate scenario assessed the hazards from explosion of the storage tank at Dow SDO. LME also assessed the hazards from an explosion of 50,000 lb (22,680 kg) of TNT on local roads. Although this quantity is legally permissible on U.S. roads, such a large quantity of TNT is not stored or used in the neighborhood, as given in PSAR table 2.2-2. Therefore, the NRC staff finds that LME has appropriately selected bounding scenarios to assess explosion (detonation) hazards from nearby facilities and transportation routes. In addition, based on the results from the ten bounding scenarios, the NRC staff finds, as discussed in PSAR section 2.2.3.2 and PSAR table 2.2-13, that overpressures generated from detonation of flammable chemicals stored at nearby facilities or transported via roads or rail lines, including BLEVE, would not develop damaging overpressure.

The NRC staff finds that LME has appropriately identified and assessed the potential hazard of immediate ignition of released flammable liquids and gaseous fuels and determined that the risk is very low and, therefore, has satisfied the requirements of 10 CFR 100.20(b) with respect to that hazard. In addition, the NRC staff finds that LME evaluated the potential hazard of immediate ignition of released flammable liquids and gaseous fuels associated with nearby facilities and transportation routes and showed that the hazard does not pose undue risk to the

proposed facility and, therefore, has satisfied the requirements of 10 CFR 100.21(e) with respect to that hazard. Additionally, as LME has conducted a safety assessment of the proposed site and facility and showed that the potential for accidents that could result in the release of significant quantities of radioactive fission products is extremely low with respect to the potential immediate ignition of released flammable liquids and gaseous fuels, the NRC staff finds that the requirements of 10 CFR 50.34(a)(1)(ii) have been satisfied with respect to that hazard.

2.2.4 Explosion (Deflagration) from Delayed Ignition of Vapor Clouds

Flammable materials may be stored and/or transported as gas or liquid under pressure near the proposed site. Accidental release of lighter than air flammable materials to the atmosphere can develop a vapor cloud (plume of chemicals floating in the air), which may travel toward the LMGS NI under prevailing atmospheric conditions. Structures in the NI area of the proposed facility may provide some confinement to the vapor cloud and may also provide the ignition source(s) for the plume to ignite. Reactors and other buildings on the NI may also provide the necessary congestion in the propagation path of the flame propagation in the vapor cloud to transition from a deflagration (rapid burning) event to a detonation (flame speed higher than the speed of sound) event. A typically gaseous chemical stored as a compressed liquid could flash to vapor isoentropically and the chemicals that do not flash to vapor would form a pool near the point of release, similar to chemicals that are liquid at standard conditions. The liquid from the pool can evaporate with time forming a vapor cloud.

The vapor cloud of a released flammable chemical can ignite after encountering an ignition source (delayed ignition) and explode (vapor cloud explosion (VCE)), if sufficient quantity of the material in the vapor cloud is within its flammable/explosive range, that is, between the LEL and the UEL. The vapor cloud may burn as a flash fire depending on the ignition source. If a pipeline carrying pressurized flammable material ruptures, the pressurized chemical can be released to the atmosphere with large momentum (like a jet). A jet fire may ensue near the release point of the chemical. The plume of vapor can also burn as a flash fire upon encountering an ignition source. A BLEVE event can take place if a tank of flammable material is exposed to fire. Blast overpressure can develop from a VCE and a BLEVE. A flash fire and a jet fire can damage nearby structures through thermal radiation although a flash fire typically has low heat load to cause any significant damage. A BLEVE can also produce a thermal flux causing damage to nearby structures.

LME assessed the damaging effects from potential VCEs, jet fires, and BLEVEs of the flammable chemicals stored at nearby facilities and/or transported on nearby transportation routes on the proposed facility structures. LME's assessment is described in PSAR section 2.2.3.3. The NRC staff reviewed the discussion in PSAR section 2.2.3.3 and audited calculation packages with additional details on the methodologies used, scenarios considered, assumptions made about the scenarios, and results of hazard analyses for explosions (deflagrations) from delayed ignition of vapor clouds.

2.2.4.1 Technical Evaluation

As stated in PSAR section 2.2.3.3, LME used the concentration of the released hazardous chemical in the vapor cloud to determine whether a release will ultimately develop into a potential explosion hazard (overpressure). If the estimated concentration of the flammable chemical at the proposed NI area falls below the respective LEL of the chemical, then it was

established that an explosion of the vapor cloud cannot take place generating an overpressure hazard. The NRC staff finds this criterion to screen overpressure hazard resulting from a postulated release of a chemical is appropriate as it is consistent with RG 1.91.

PSAR section 2.2.3.3 also states that LME assumed the chemicals typically stored as gas would be released in a single puff and that dispersion of a chemical in the vapor cloud was modeled using NUREG–0570 and RG 1.145. LME modeled meander of the plume using meandering coefficients, as suggested by RG 1.145. The NRC staff finds this acceptable as modeling follows appropriate guidance.

Flammable Vapor Clouds from Dow SDO

As stated in PSAR section 2.2.3.3.1, flammable chemicals are stored at the Dow SDO facility at a distance of 3,000 ft to 6,864 ft (914 m to 2092 m) from the proposed LMGS NI. The NRC staff's assessment of hazardous chemicals released accidentally in the rail yards is discussed below under hazards from rail lines. PSAR table 2.2-2 lists potential explosive and toxic chemicals near the proposed LMGS site and PSAR table 2.2-10 lists the chemicals stored at Dow SDO assessed as potential hazards from flammable vapor clouds. LME obtained the location of each chemical stored in the Dow SDO facility. LME assumed the potential to pose a hazard deterministically by estimating the chemical concentration at the proposed NI area from an external release of a chemical and comparing whether the concentration of the chemical fell below the respective LEL value in this assessment. As discussed above, the assessment involves potential dispersion and meandering of the plume in the atmosphere while floating away from the point of release. LME has used the atmospheric stability class and wind speed combinations occurring more than 95% of the time.

PSAR section 2.2.3.3.1 states that hazards associated with the flammable vapor clouds of all chemicals stored at Dow SDO ~~chemicals~~ are acceptable, as shown in PSAR table 2.2-13, except butene. The NRC staff finds that the concentration of all chemicals except butene at Dow SDO would fall below the respective LEL value in a VCE, as shown in PSAR table 2.2-13. Chemicals undergoing an immediate explosion (detonation) are reviewed in SE section 2.2.3. PSAR section 2.2.3.3.1 states that hazards associated with a release of approximately 250,000 lb (113,400 kg) of butene at Dow SDO was evaluated probabilistically using a Monte Carlo analysis. The hazard associated with the release of butene becomes a rare event if its occurrence can be shown to be less than 1×10^{-6} per year, following the guidance in section 2.2.3 of NUREG–0800. A facility need not consider a rare event as a design-basis event. This probabilistic approach is acceptable to the NRC staff as it follows the guidance in RG 1.91 and NUREG–0800, section 2.2.3.

As butene is stored at Dow SDO in storage tanks, LME used the rate of failure of single-walled storage tanks from the failure database maintained by the Government of the Netherlands to model the rate of tank failure resulting in release of the butene. Based on the database, single-walled tanks fail at a rate of 5×10^{-6} per year (RIVM, 2008). LME assumed that butene is stored in a single-containment atmospheric tank at Dow SDO. The NRC staff finds this assumption acceptable because an outer shell may or may not be present at the butene tank at the Dow SDO; and, even if an outer shell is present, it would not provide any containment. This tank category (i.e., single-walled storage tanks) has the highest failure rate (5×10^{-6} per year), as listed in RIVM (2008) table 3.5. If present, a protective tank outer shell could reduce this failure rate by at least an order of magnitude; however, it is conservative to assume no outer shell.

Therefore, the NRC staff concludes that assumptions made regarding butene tank failure are conservative.

The conditional probability that released butene will be directly ignited (i.e., immediate ignition) is 0.7 from the failure database maintained by the Government of the Netherlands. The probability that the flammable vapor cloud will ignite after floating downwind (i.e., delayed ignition) is 0.4 from the same database. The NRC staff notes that delayed ignition is possible only if the plume was not immediately ignited near the release point so that a plume can form and float away.

LME conducted 20,000 realizations of the atmospheric parameters: wind speed, wind direction, and stability class. Out of 20,000 realizations, four generated concentrations above the LEL of butene at the proposed LMGS NI, giving a rate of 3.6×10^{-4} hazardous conditions per release with 95 percent confidence. Therefore, combining the estimated tank failure frequency with the estimated rate of generating a chemical concentration above the LEL, LME estimated that release of butene from a tank at Dow SDO would produce hazardous conditions with a frequency of 2.2×10^{-10} per year. As there are 27 sources of butene at the Dow SDO facility, as stated in PSAR section 2.2.3.3.1, LME estimated the total release of butene resulting from either leak or catastrophic failure of the storage tanks at Dow SDO would be 5.9×10^{-9} per year with 95% confidence at the LMGS NI area.

The NRC staff finds the estimated annual frequency is significantly smaller than the threshold value to be a design-basis hazard, based on guidance in section 2.2.3 of NUREG–0800, and concludes that potential butene release from storage tanks at Dow SDO will not be a credible hazard to the proposed facility.

Flammable Vapor Clouds from Other Facilities

PSAR section 2.2.3.3.2 identified three facilities within 5 mi (8 km) of the proposed site storing chemicals capable of forming flammable vapor clouds: (1) Calhoun Chemical, (2) Seadrift Coke, and (3) INEOS Nitriles. Although these facilities use several chemicals, as listed in PSAR table 2.2-2, PSAR table 2.2-10 lists only the chemicals at these facilities that can undergo a VCE. The NRC staff finds that a radius of 5 mi (8 km) from the proposed NI is an appropriate distance to consider flammable vapor clouds from other facilities as a potential hazard because atmospheric dispersion is expected to significantly dilute the flammable chemical concentration in the vapor cloud as the cloud travels and because this approach follows RG 1.78, Revision 2.

As shown in PSAR table 2.2-13, the Calhoun Chemical facility is 3.5 mi (5.6 km) from the proposed site and stores isoamyl acetate and Mr. Solvent (Petroleum Raffinates). The concentration of these chemicals in a vapor cloud at the proposed LMGS site was estimated assuming the most stable atmospheric stability class G and very low wind speed. The concentration of these chemicals at the LMGS NI based on the release of the entire amount of these chemicals from storage tanks at Calhoun Chemical was estimated to be below the respective LEL value, as described in PSAR section 2.2.3.3.2. Consequently, PSAR section 2.2.3.3.2 states release of all chemicals stored at Calhoun Chemicals would not develop flammable vapor clouds at the proposed LMGS NI area. The NRC staff finds that flammable hazardous chemicals stored at the Calhoun Chemical facility would not pose a credible VCE hazard as the chemical concentration in the plume will fall below the respective LEL.

As shown in PSAR table 2.2-13, the Seadrift Coke facility is 1.9 mi (3.1 km) from the proposed site and stores petroleum naphtha. As deterministic analysis shows that the release of

10,000,000 lb (4,536,000 kg) of petroleum naphtha at Seadrift Coke can become a credible hazard to the proposed LMGS facility, LME conducted a probabilistic analysis to estimate the frequency of occurrence. The Monte Carlo analysis sampled atmospheric parameters, such as wind speed, wind direction, and stability class similar to that discussed before with Dow SDO. Out of 20,000 realizations, 19 resulted in a chemical concentration above the LEL of petroleum naphtha at the proposed LMGs NI, giving a rate of 1.2×10^{-3} hazardous conditions per release with 95% confidence. Assuming that petroleum naphtha is stored only in single-walled tanks in Seadrift Coke, the frequency of hazardous explosions is estimated to be 7.2×10^{-10} per year with 95% confidence. The NRC staff finds that LME used the approach to assess potential hazards from failure of storage tanks storing petroleum at Seadrift Coke similar to that used for tanks at Dow SDO. In addition, the estimated annual frequency of hazardous explosions is well below the threshold given in section 2.2.3 of NUREG–0800 to become a design-basis event. Therefore, the NRC staff finds that a potential failure of the tanks storing petroleum naphtha at Seadrift Coke would not be a credible hazard to the proposed facility.

LME estimated the concentration of each flammable chemical stored at INEOS Nitriles listed in PSAR table 2.2-10 at the proposed LMGS NI. The release of the entire quantity of each chemical stored at the facility was assumed. As stated in PSAR section 2.2.3.3.2, all chemicals except propylene produced acceptable conditions under the worst atmospheric conditions (extremely stable atmosphere with little wind). Therefore, LME conducted a probabilistic analysis to estimate the frequency of occurrence for the release of propylene.

LME assessed the hazard posed by release of the entire 2,655,000 lb (1,204,000 kg) of propylene stored at INEOS Nitriles to the proposed LMGS NI using Monte Carlo analysis, as discussed in PSAR section 2.2.3.3.2. Out of 20,000 realizations, 13 resulted in concentrations above the LEL of propylene giving the rate of explosion of 9.5×10^{-4} per release with 95% confidence. Assuming propylene is stored at INEOS Nitriles in single-walled tanks, the frequency of propylene release at INEOS Nitriles resulting in a VCE at the proposed LMGS NI is estimated to be 5.7×10^{-10} per year, as stated in PSAR section 2.2.3.3.2. As INEOS Nitriles stores propylene in three large tanks, the frequency of hazardous explosion is estimated to be 1.7×10^{-9} per year with 95% confidence. As this annual frequency is smaller than compared to that recommended in section 2.2.3 of NUREG–0800, LME concluded in PSAR section 2.3.3.2.2 that propylene tank failure at INEOS Nitriles will not pose a credible hazard to the proposed LMGS.

The NRC staff finds that LME has assumed propylene is stored at INEOS Nitriles in single-walled tanks and, therefore, has taken the maximum probability of tank rupture resulting in a release to the surrounding area. Using Monte Carlo analysis, LME estimated the conditional probability that the propylene plume would still be flammable when it reaches the proposed NI area. The NRC staff also finds that the estimated annual frequency of propylene VCE indicates that it is not a design-basis hazard to the proposed LMGS.

The NRC staff has reviewed PSAR section 2.2.3.3.2 and examined the analyses of potential hazards caused by flammable vapor clouds from other facilities. As the plume dispersion will be affected by wind speed, wind direction, and atmospheric stability class, only these parameters were randomly sampled from their given site-specific distribution. The plume is allowed to disperse under the selected atmospheric conditions in each Monte Carlo run. Using meander coefficients suggested in RG 1.145, the chemical concentration in the plume is estimated at the proposed NI location and compared with its LEL value for potential for explosion. Using

probabilistic analysis, LME showed that total release of all the chemicals in storage at the nearby facilities developing a VCE at the proposed NI area is a rare event, compared to the value recommended in section 2.2.3 of NUREG–0800. Therefore, the NRC staff finds that flammable chemicals stored at nearby facilities do not pose a credible hazard to the proposed facility from explosion of vapor clouds.

Flammable Vapor Clouds from Roadway Transport

As stated in PSAR section 2.2.3.3.3, LME evaluated all chemicals used in the vicinity of the proposed LMGS assuming transportation on local roadways using trucks, except chemicals at Dow SDO. As discussed in PSAR section 2.3.3.3.1, chemicals at the Dow SDO facility are located 1,300 ft to 1.3 mi (0.4 km to 2.1 km) away from the proposed LMGS site, which is closer than the nearest roadway. LME evaluated the flammable vapor clouds from chemicals at Dow SDO, taking the closer of the actual distance and the distance of proposed LMGS site to the nearest roadway TX-35, which is 0.81 mi (1.3 km) away.

PSAR table 2.2-5 lists the hazardous chemicals that are transported on nearby roadways; however, only a subset of these chemicals can generate flammable vapor clouds. PSAR table 2.2-11 lists the chemicals LME evaluated for the potential to develop vapor cloud explosions while transported by trucks. LME estimated the concentration of each of these chemicals at the proposed LMGS NI assuming total release of each chemical. As previously discussed, LME assumed that the total release is the smaller of 50,000 lb (22,680 kg), based on RG 1.91, or the maximum quantity used at any facility within 5 mi (8 km) of the proposed site. The NRC staff finds this assumption reasonable because it follows RG 1.91.

PSAR section 2.2.3.3.3 states that none of the chemicals listed in PSAR table 2.2-11 generates a vapor cloud with concentration higher than the respective LEL at the proposed LMGS NI for stability class-wind speed combinations that occur more than 95 percent of the time. Therefore, based on the use of acceptable assumptions, an acceptable evaluation method, and resulting chemical concentrations below the respective LEL at the proposed site, the NRC staff finds that hazardous chemicals transported by trucks near the proposed LMGS site will not result in a credible flammable vapor cloud hazard as the cloud will not explode.

Flammable Vapor Clouds from Rail Transport

PSAR section 2.2.3.3.5 describes the assessment of potential hazards at the proposed LMGS site from an accidental release of flammable chemicals from railcars while they are being moved around at the Dow SDO railyard or traveling on either the UP or BNSF line. As stated in PSAR section 2.2.3.3.5, LME received proprietary information on shipment frequencies and movement in the railyard from Dow SDO, UP, and BNSF and used the information to assess the potential hazards to the proposed facility. As discussed above, both the UP and BNSF tracks are approximately 1.4 mi (2.3 km) from the proposed site, and the railyard is approximately 0.3 mi (0.5 km) away from the proposed LMGS NI.

In PSAR section 2.2.3.3.5, LME identifies the chemicals from PSAR table 2.2-5 that can potentially explode if accidentally released from the railcars as (1) ammonia, (2) acrylonitrile, (3) butylene, (4) ethylene oxide, (5) isopentane, (6) propylene oxide, and (7) solids (chemicals that can explode when aerosolized as dust).

PSAR section 2.2.3.3.5 states that the estimated concentration in the vapor cloud at the proposed NI location of all chemicals released from railcars on the UP or BNSF lines would be below the respective LEL. Therefore, LME concluded that the hazards from flammable chemicals while being transported on either the UP or BNSF line are not credible hazards to the proposed facility. In addition, PSAR section 2.2.3.3.5 states that release of ammonia, acrylonitrile, and propylene oxide from railcars being moved around at the Dow SDO railyard would not be hazardous to the proposed facility as the estimated concentration at the proposed NI location is below the respective LEL. The NRC staff agrees with the results as the released chemicals concentration in air falls below the threshold to ignite upon encountering an ignition source.

PSAR section 2.2.3.2 identified solids as potentially explosive when aerosolized. As discussed in section 2.3.3 of this SE, solids were evaluated in Case 10 for potential to develop more than 1 psid (6.9 kPa) overpressure. At a distance of 1,290 ft (393 m), the overpressure from detonation is 1 psid (6.9 kPa). A maximum of 10% of the allowable railcar capacity is assumed to be aerosolized. Because the mass floating in air would continuously decrease as larger particles fall to the ground under gravity and the available distance from the railyard is larger than the 0.24 mi (0.4 km) determined from detonation, the NRC staff finds potential explosion hazards at the proposed NI from aerosolized solid particles is not credible.

PSAR section 2.2.3.3.5 identifies potential releases of butylene, ethylene oxide, and isopentane from railcars as resulting in concentrations higher than their respective LEL concentration and, therefore, pose a potential explosion hazard. To address these hazards, LME conducted a probabilistic assessment to determine the annual frequency of explosion hazards at the proposed NI from the vapor clouds in two different scenarios:

1. Hazardous chemicals in movement: Railcars are moved around in the rail yard such that the release of the chemical will be the result of an accident releasing the entire hazardous content.

2. Stationary hazardous materials: Railcars loaded with hazardous chemicals are used as temporary storage.

LME assumed that every movement in the railyard would including traveling the entire length of the yard, which is approximately 5.5 mi (8.8 km) long. The NRC staff notes that LME assumed the entire length of the railyard as the exposure distance following RG 1.91 and assumed a release within this 5.5 mi (8.8 km) of the railyard can pose an explosion hazard at the proposed NI.

In PSAR section 2.2.3.3.5, LME estimated the frequency of significant release from railcars while being moved in the Dow SDO railyard as 6.75×10^{-8} per railcar-mile. LME used the accident rate of rail transportation within a yard of 3.0×10^{-6} per car-mile from (Federal Emergency Management Agency (FEMA),1990) and modified that rate with the probability of complete cargo loss (FEMA, 1990) to develop the crash rate. LME assumed that the spill probability in a yard is half of that on a main line because of low-speed railcar movements. The NRC staff finds that the estimated crash rate is acceptable because it was developed using data from an authoritative source.

In addition, LME assumed, using statistical information from the Government of the Netherlands (RIVM, 2008), that immediate ignition of gases stored as liquid takes place for 80% of the spills. Consequently, in 20% of the spills, the material will flash to gas upon release, and the vapor will float away. The gaseous material in these vapor clouds may undergo delayed ignition upon encountering an ignition source and detonate, generating a high overpressure. The Government of the Netherlands (RIVM, 2009) states that a high overpressure generates from detonation in 40% of the spills that flash to gas only. The vapor cloud will deflagrate the rest (60%) of the time.

Using Monte Carlo methods, LME estimated probabilistically the fraction of the chemical releases at the Dow SDO railyard resulting in a vapor cloud at the proposed NI area with the chemical concentration above its LEL, by varying the location of the release point along the 5.5-mi (8.8 km) long track in the railyard, and a combination of wind speed, wind direction, and stability class. The Monte Carlo analysis selected a random point of release of a chemical along 5.5 mi (8.8 km) of the rail yard. In addition, the vapor cloud formed as a result of the release will float away under prevailing weather conditions, represented by the wind speed, wind direction, and stability class. As stated in PSAR section 2.2.3.3.5, the estimated annual frequency was multiplied by the total number of railcar movements taking place annually.

As reported in PSAR section 2.2.3.3.5, the combined frequency of VCE at the proposed NI from railcar movements carrying butylene, ethylene oxide, and isopentane in the Dow SDO railyard is 4.82×10^{-7} annually with 95% confidence.

As stated in PSAR section 2.2.3.3.5, LME also analyzed the scenario in which the railyard is used for temporary storage of hazardous chemicals. The railyard has capacity to store approximately 2,000 railcars. Although the railcars parked in the railyard may be empty, LME assumed chemicals are permanently stored at the railyard and that the railcars are completely full of butylene, ethylene oxide, and isopentane. As given in PSAR section 2.2.3.3.5, the combined frequency of railcars storing these three chemicals generating an explosion hazard at the proposed NI area is 2.2×10^{-7} explosions per year with 95% confidence.

LME stated in PSAR section 2.2.3.3.5 that the combined hazard from both stationary and moving hazardous chemicals in the Dow SDO railyard is 7.02×10^{-7} explosions per year with 95% confidence. Using section 2.2.3 of NUREG–0800, the NRC staff notes that rail cargo causing explosion hazards at the proposed NI is below the threshold annual frequency of 1.0×10^{-6} per year to become a design-basis hazard. The NRC staff finds a higher frequency acceptable because LME evaluated the potential explosion hazards at the rail yard from chemicals stored in railcars using conservative assumptions. For example, LME assumed that a railcar is always filled at full capacity and the yard always stores 2,000 railcars full of these hazardous chemicals. In reality, the number of rail cars in the yard could be less and all those rail cars may not be completely full of hazardous chemicals.

In summary, the NRC staff finds that the potential explosion hazards from rail transport of hazardous chemicals would not be a design-basis hazard because the estimated concentration of all the chemicals in the vapor cloud at the proposed NI location would be below the respective LEL values. The NRC staff also finds that the potential hazard of chemicals exploding in rail cars while at the rail yard is less than the threshold frequency and, therefore, is also not a design-basis hazard for the proposed facility.

Flammable Vapor Clouds from Barge Transport

PSAR section 2.2.3.3.4 discusses the potential explosion hazards to the proposed LMGS NI from cargoes transported by barges on the Victoria Barge Canal (VBC). PSAR table 2.2-11 lists the flammable chemicals transported by barges on the VBC that require an evaluation for potential explosion hazard from vapor clouds.

LME found that release of the entire amount of acetone and cyclohexane shipments using barges in the VBC would not pose a credible hazard to the LMGS NI as the estimated concentrations of both chemicals are below the respective LELs. However, as butadiene produced a concentration above its LEL, LME assessed its potential to become a credible hazard probabilistically. LME analyzed the release of the entire 3,720,000 lb (1,687,000 kg) of butadiene corresponding to the maximum mass per barge on the VBC. LME also assumed 120 such shipments take place annually, as stated in PSAR section 2.2.3.3.4. As discussed above, LME collected this information from the Port of Victoria as well as the industrial facilities with docks on the canal.

LME randomly sampled from distributions of the wind speed, wind direction, and atmospheric stability class. In addition, following RG 1.91, LME also sampled the release location on the VBC within 5 mi (8 km) of the proposed LMGS site. LME estimated this exposure length of the VBC to be 9.55 mi (15.4 km), following RG 1.91. As the hazard being analyzed is a vapor cloud of butadiene released on the VBC floating to the proposed NI area and exploding, the NRC staff finds that, as provided in RG 1.78, chemicals stored or situated at distances greater than 5 mi (8 km) need not be considered.

As stated in PSAR section 2.2.3.3.4, out of 100,000 realizations, 30 realizations of butadiene release resulted in concentrations above the LEL at the proposed NI translating to a rate of 3.9×10^{-4} explosions per release with 95% confidence. LME used release statistics from FEMA (1990) to estimate total loss cargo in barge grounding events as the scenario called for total loss of butadiene while being transported on the VBC by barge. Using FEMA (1990), the rate of a vessel releasing its entire chemical inventory is modeled as 1.5×10^{-6} per mile. The NRC staff finds the use of FEMA (1990) for release statistics because it is a widely used, authoritative source on accidents involving transport of hazardous materials.

Using information from the Government of the Netherlands, LME assumed that a released butadiene vapor cloud ignites immediately (direct ignition) 70% of the time. Additionally, LME assumed the probability that a vapor cloud will be ignited away from the release point (delayed ignition) is 40% (RIVM, 2009). The NRC staff notes a delayed ignition of the vapor cloud is possible only if a direct or immediate ignition of the cloud did not take place. The NRC staff finds the use of failure information from the Government of the Netherlands acceptable because it is widely used in quantitative risk analysis in the chemical industries worldwide and includes the chemicals evaluated as potential hazards to the LMGS site.

As given in PSAR section 2.2.3.3.4, the frequency of hazardous explosions of butadiene on barges transiting the VBC is 8.1×10^{-8} accidents per year with a 95% confidence. The NRC staff finds that release of butadiene, acetone, and cyclohexane while being transported in barges on the VBC would not pose a credible hazard to the proposed facility because the estimated concentration of each chemical is below the respective LEL or the annual frequency of an explosion is smaller than that specified in section 2.2.3 of NUREG–0800 for the chemical.

Flammable Vapor Clouds from Pipelines

PSAR table 2.2-3 lists pipelines carrying gaseous materials within 5 mi (8 km) of the proposed LMGS site. PSAR table 2.2-4 lists pipelines carrying materials in liquid form. The hazards posed by these pipelines to the proposed LMGS site are the sum of hazards posed by individual pipelines. LME used both deterministic and probabilistic methods to evaluate the potential hazards from flammable material released from these pipelines and considered both a leak and a complete rupture of a pipeline. The NRC staff notes that a complete rupture of a pipeline typically bounds a leak.

As stated in PSAR section 2.2.3.3.6, LME considered 12 bounding pipeline release scenarios to assess the hazards posed by a VCE. PSAR table 2.2-12 lists these scenarios. The first three scenarios are leaks from a 1-in. (2.5-cm) diameter hole developing in the pipelines carrying propane, ethylene, and propylene located approximately 0.5 mi (0.8 km) from the facility. As these pipelines are close to the proposed facility, a leak in the pipelines may affect the proposed facility. The estimated concentration of each chemical at the LMGS NI using a deterministic method is below the LEL of the chemical. Therefore, the vapor cloud cannot be ignited when it is over the proposed facility. Consequently, a vapor cloud would not pose an explosion hazard to the proposed facility.

For the remaining nine scenarios, LME assumed that the pipeline suffers a catastrophic failure resulting in a guillotine break. The NRC staff notes that release of chemicals through a leak is much more common than a complete full-bore break. For scenarios 8 and 12, LME found that the two 8-in. (20-cm) diameter pipelines operated by INEOS that carry ammonia and acrylonitrile do not pose a credible hazard because the concentration at the proposed NI is below the respective LEL values. Similarly, for scenario 9, LME found that the 4-in. (10-cm) diameter ethylene pipeline on the west side of Dow SDO does not pose a credible hazard for the same reason.

Scenario 11 involves complete rupture of the 30-in. (76-cm) diameter natural gas (taken as composed of only methane) pipeline operated by Enterprise Products Operating LLC. Because of its size and proximity (0.59 mi) to the proposed LMGS site, this scenario bounds all other nearby methane/natural gas pipelines and gas wells. LME evaluated this scenario with different release rates from the pipeline and estimated the methane concentration in the plume as it floats toward the proposed facility. As methane is a buoyant gas, the plume rises in the atmosphere as it floats away. LME estimated the methane concentration considering the dispersion and meander of the plume, following RG 1.145, and determined by trial and error that the optimum release rate would be 861 lb/sec (391 kg/sec) from the 30-in. (76-cm) diameter natural gas pipeline, as the plume contains maximum amount of methane. This release rate is smaller than the release rate possible from a complete rupture of the 30-in.(76-cm) diameter natural gas pipeline.

LME stated in PSAR section 2.2.3.3.5 that a release rate higher than 861 lb/sec (391 kg/sec) from a rupture of the pipeline would result in the plume rising to higher altitudes. Consequently, when this plume reaches the proposed NI area, a major portion of the plume will be above the structures. However, at those elevations, there will not be any structures to provide the confinement necessary for explosion. In addition, there will not be any ignition sources at that elevation to initiate the explosion. On the other hand, release of natural gas at lower rates than 861 lb/sec (391 kg/sec) will disperse methane in the atmosphere with significantly diluted

concentrations when the plume reaches the proposed NI area and, consequently, there will not be an explosion as the concentration would be below the LEL.

In addition, methane concentration even at a release rate of 861 lb/sec (391 kg/s) falls below the LEL when the vapor cloud reaches the proposed site, as stated in PSAR section 2.2.3.3.6. Therefore, the NRC staff finds that a release from the nearby 30-in. (76-cm) diameter natural gas pipeline will not pose a VCE hazard affecting the proposed facility. This conclusion also bounds any methane release from nearby gas wells because of the size and proximity (0.59 mi (0.95 km)) to the proposed LMGS site of the pipeline, as explained above.

LME addressed the remaining pipeline-related scenarios probabilistically, as stated in PSAR section 2.2.3.3.6. LME estimated the frequency of rupture of a pipeline leading to delayed ignition of the vapor cloud resulting in a detonation event. The NRC staff notes that a vapor cloud needs some confinement and adequate congestion in its propagation path to explode in a detonation; otherwise, the vapor cloud will burn rapidly (deflagration mode). Consequently, assuming detonation occurs for a given plume of vapor is conservative for assessing structural damage. The NRC staff also notes detonation of a given volume of vapor would produce larger peak pressure than a deflagration event. The NRC staff notes the structural damage is correlated with the peak pressure, as discussed in RG 1.91.

LME used the statistical information from the Netherlands (RIVM, 2005) to estimate the annual frequency of hazardous explosions from pipeline rupture, as described in PSAR section 2.2.3.3.6. A large release of flammable gas is assumed to ignite at the release point (e.g., forming a jet fire and/or flash fire) 70% of the time, and the flammable vapor clouds are assumed to reach the proposed NI the remaining 30% of the time. The NRC staff finds these assumptions acceptable because these statistics are conservative as they ignore those cases where ignition does not take place at all.

Further, the NRC staff independently assessed the conservativeness of the direct ignition probability value based on other sources, such as Acton and Baldwin (2002) and Acton et al. (2016), who used the worldwide pipeline failure data and associated ignition data for the period 1970 through 2014, maintained by an international group of transmission pipeline companies. This dataset also included U.S. operational experience available from the Pipeline and Hazardous Materials Safety Administration of the U.S. Department of Transportation (USDOT) for the period of 2002 through 2014. Acton and Baldwin (2008) and Acton et al. (2016) found a statistical correlation between the probability of direct ignition and the operating pressure and diameter of a pipeline. The NRC staff used the empirical equation, developed by Acton et al. (2016), to estimate the conditional probability of direct ignition of the plume released rupture of the Agua Dulce-Sabine Line Natural gas pipeline. The NRC staff calculates that the probability of direct ignition of this pipeline is 0.58. Consequently, the NRC staff finds that LME has used a conservative probability value to assess the hazard near the proposed facility.

The NRC staff also expects that the pipelines will be located below ground (at least 30 in. (76 cm) or more earth cover) near the proposed facility following the USDOT regulations (49 CFR 192.327). Therefore, a release from the buried pipeline would be a momentum-dominated lighter than air release. As a result, availability of flammable gas at the ground level would be limited, as discussed by Acton et al. (2016). Based on PSAR figure 2.2-3, the possibility of finding sources capable of directly igniting the released gas near the proposed facility is expected to be low.

Based on RIVM (2009), the flammable vapor cloud will explode after reaching the proposed NI 40% of the time. LME calculated a rate of hazardous explosion to be 6.78×10^{-7} per mile-year in PSAR section 2.2.3.3.6. The NRC staff finds this rate acceptable because it was developed using information from authoritative sources.

For scenarios 4 and 10, LME estimated the exposure length of two adjacent propane pipelines operated by Dow SDO at a distance of 0.43 mi (0.69 km) from the proposed site to be 6.21 mi (9.99 km) in PSAR section 2.2.3.3.6. Using similar assumptions as above, LME used Monte Carlo analysis and estimated a probability of 2.01×10^{-3} flammable vapor clouds (31 out of 20,000 realizations) at the proposed NI with 95% confidence. The total hazard frequency is estimated to be 1.69×10^{-8} per year with 95% confidence, as given in PSAR section 2.2.3.3.6. For scenario 5, the ethylene pipeline operated by Dow SDO runs in the same lane with the propane pipeline. LME estimated the hazard from complete rupture of this ethylene pipeline using Monte Carlo analysis. Out of 20,000 realizations, 129 developed a flammable concentration in the proposed NI area, which translates to a frequency of flammable vapor cloud at the proposed NI of 7.38×10^{-3} flammable vapor clouds per release with 95% confidence. Following a similar procedure used for propane pipelines, the hazard frequency is estimated to be 3.11×10^{-8} per year, as given in PSAR section 2.2.3.3.6.

Scenarios 6 and 7 evaluate a propylene pipeline and a propane pipeline 0.59 mi (0.95 km) north of the proposed LMGS site listed in PSAR table 2.2-12. LME estimated the exposure distance to be 10 mi (16 km) within the 5-mi (8 km) radius of the proposed site. LME estimated the frequency of flammable vapor clouds using Monte Carlo analysis and estimated 4.34×10^{-4} (5 out of 20,000 realizations) and 1.13×10^{-3} (16 out of 20,000 realizations) flammable vapor clouds per year of propylene and propane, respectively, with 95% confidence. As stated in PSAR section 2.2.3.3.6, rupture of the propylene pipeline results in a hazard frequency of 2.95×10^{-9} per year, following similar arguments of delayed ignition. Rupture of the propane pipeline results in a hazard frequency of 7.67×10^{-9} per year assuming delayed ignition.

Scenario 10 in PSAR table 2.2-12 evaluates a propane pipeline south and west of the proposed site at a distance of 0.9 mi (1.4 km). LME estimated the exposure distance of this pipeline to be 4.34 mi (6.98 km). Monte Carlo analysis was used to estimate the frequency of flammable concentration at the proposed NI to be 2.16×10^{-4} (2 out of 20,000 realizations) per release with 95% confidence. Following the same procedure as other pipeline evaluations, LME estimated the hazard frequency to be 6.36×10^{-10} per year, as stated in PSAR section 2.2.3.3.6.

LME has added these individual hazard frequencies of developing a flammable vapor cloud at the proposed NI to develop the cumulative frequency of 5.9×10^{-8} per year with 95% confidence. This is acceptable because the VCEs are coming from 12 different sources.

The NRC staff notes that the hazard estimation process used by LME has conservatism. For example, any release of a hazardous chemical from a pipeline is assumed to have sufficient quantity to form an ignitable vapor cloud. A leak may have smaller amounts of chemicals available for ignition. In addition, any loss of chemicals through absorption in soils near the rupture/leak location is assumed to be zero. Additionally, the analyses assumed that ignition sources of sufficient strength would be available at the proposed NI area. Different chemicals require different amounts of energy to ignite due to varying reactivity. For example, methane (natural gas) requires significantly higher energy to ignite compared to other chemicals such as propane and ethylene. Therefore, the NRC staff finds that the cumulative hazard frequency of

5.9×10^{-8} per year with 95% confidence is well below the annual frequency suggested in section 2.2.3 of NUREG–0800. The NRC staff further concludes that formation of flammable vapor clouds from pipeline ruptures (as well as leaks) does not pose a credible hazard to the proposed facility.

Flammable Vapor Clouds from Oil/Gas Wells

LME discussed in PSAR section 2.2.2.3 active and potentially active oil and gas wells within 5 mi (8 km) of the proposed site and lists them in PSAR table 2.2-9. The nearest gas well is a 2-in. (5-cm) diameter well located 2.1 mi (3.4 km), and the nearest oil well is a 2-in. (5-cm) diameter well located 4.1 mi (6.6 km) away from the proposed site. Flaring of excess gas is routinely conducted in gas wells as a part of the production process. Potential hazard to the proposed facility includes natural gas leaking out of a gas well and forming a vapor cloud. This vapor cloud may float toward the proposed site and ignite, developing an overpressure. Alternatively, flared natural gas may develop significant thermal loads for the proposed facility. The NRC staff finds that these wells are too far away from the site to generate a concentration above the methane LEL near the proposed site, as shown in the analysis of a potential hazard from a rupture of the 30-in. (76-cm) diameter natural gas pipeline. In addition, the wells are too far to impose any appreciable thermal hazard to the proposed facility from gas flaring.

Therefore, based on the above discussion, the NRC staff finds that these wells will not produce significant overpressure or thermal radiation hazards to affect safe operation at the proposed facility.

2.2.4.2 Conclusion

The NRC staff finds that LME has appropriately identified and assessed the potential VCE hazard and determined that the risk is very low and, therefore, has satisfied the requirements of 10 CFR 100.20(b) with respect to that hazard. In addition, the NRC staff finds that LME evaluated the potential VCE hazard associated with nearby facilities and transportation routes and showed that the hazard does not pose undue risk to the proposed facility and, therefore, has satisfied the requirements of 10 CFR 100.21(e) with respect to that hazard. Additionally, as LME has conducted a safety assessment of the proposed site and facility and showed that the potential for accidents that could result in the release of significant quantities of radioactive fission products is extremely low with respect to a potential VCE, the NRC staff finds that the requirements of 10 CFR 50.34(a)(1)(ii) have been satisfied with respect to that hazard.

2.2.5 Fires

LME evaluated the thermal loads on the SSCs of the proposed facility that would result from a wildfire (or fire of natural vegetation) and chemical fires and compared the heat load generated to the heat load and its duration limits with those given in NUREG/CR–3330 “Vulnerability of Nuclear Power Plant Structures to Large External Fires” (ML062260290) for assessing potential thermal hazards to SR SSCs.

2.2.5.1 Technical Evaluation

PSAR section 2.2.3.4 states that the land surrounding the proposed site is either crop fields or Dow SDO and that there are no wooded areas nearby. Consequently, a vegetation fire that

could impact the proposed site will only include burning of surrounding crops, not a large fire or heat load. LME estimated the fire load from a worst-case brush fire using the fire modeling guidelines of the United States Forest Service to be at most 12.7 (kilowatt per square meter (kW/m^2))of incident heat flux lasting for 12 minutes burn duration. Results of experiments given in NUREG/CR–3330 show that this fire load would not produce a credible hazard to the proposed facility. Therefore, the NRC staff finds that LME has used acceptable assumptions to categorize the potential offsite fire danger at the proposed facility. In addition, LME used appropriate guidance to show that potential thermal hazard from offsite fires would not be a credible hazard to the proposed facility.

LME addressed the effects of chemical fires on safe operation of the proposed facility in PSAR section 2.2.3.4. LME estimated the heat flux at a distance of 1,200 ft (366 m) from the proposed facility resulting from:

1) A jet fire of ethylene released from a 12-in. (30-cm) diameter opening in a tank or the nearby pipeline

2) A pool fire following spills from tanks containing hexane, hexene, and acetone

3) A BLEVE event of chemicals stored as pressurized liquid but are gases at standard conditions.

The NRC staff finds that assuming 1,200 ft (366 m) distance from nearby facilities is conservative because PSAR section 2.2.3.3.1 identifies the nearest chemical storage at Dow SDO as approximately 3,000 ft (914 m) from the proposed LMGS site.

LME assumed that the pipeline carrying ethylene near the proposed site ruptures causing a jet fire from the release through a 12-in. (30-cm) diameter opening. This pipeline has a diameter of 12 in. (30-cm), as given in PSAR table 2.2-12 and figure 2.2-3, and is operated by Seadrift Pipeline Corporation. This pipeline is located 2,270 ft (692 m) from the proposed site, as stated in PSAR table 2.2-2.

LME estimated the maximum heat load from this ethylene jet fire to be 5.34 kW/m^2 at a distance of 1,200 ft (366 m) from the proposed site. Experimental results in NUREG/CR–3330 show that a concrete wall can withstand a heat load of 15 kW/m^2 for 11.6 hours. In addition, the heat load of 5.34 kW/m^2 was estimated at 1,200 ft (366 m) away from the proposed NI, compared to the distance to the nearest pipeline at 2,270 ft (692 m), as stated in PSAR table 2.2-12. Therefore, based on the above discussion, the NRC staff finds that a bounding jet fire originating at the nearest ethylene pipeline or at a storage tank at Dow SDO does not constitute a credible hazard and would not damage the SR structures at the proposed facility.

LME also considered pool fires of hexane, hexene, and acetone at Dow SDO. These are the most common liquid chemicals used at Dow SDO. A pool fire of 294,000 lb (133,000 kg) of hexane produced the maximum heat load of 21.9 kW/m^2 1,200 ft (366 m) from the proposed site and burned for 39 seconds. NUREG/CR–3330 suggests that a concrete wall can withstand a heat load of 25 kW/m^2 for 8.1 hr. Therefore, the NRC staff finds that the worst-case pool fire at Dow SDO does not constitute a credible hazard and would not damage the SR structures at the proposed facility.

LME also estimated the heat load from a BLEVE fireball of the largest propylene tank at Dow SDO. The propylene tank at Dow SDO can hold 294,000 lb (133,000 kg) of propylene. A BLEVE fireball produced heat load of 33.2 kW/m^2 1,200 ft (366 m) from the proposed site and burns for 17 seconds, as stated in PSAR section 2.2.3.4. NUREG/CR–3330 states that a concrete wall can withstand a heat load of 50 kW/m^2 for 4.95 hr. Therefore, the NRC staff finds that a BLEVE fireball does not constitute a credible hazard and would not damage the SR structures at the proposed facility.

2.2.5.2 Conclusion

Based on the above discussion, the NRC staff notes that LME has estimated the heat loads on the proposed facility from a worst-case jet fire, a pool fire, and a BLEVE fireball. The NRC staff also notes that these heat loads are not large enough to cause thermal damage to the proposed facility based on experimental results from a concrete wall, as given in NUREG/CR–3330. Therefore, the NRC staff finds that the SR structures will be able to withstand these thermal loads without affecting safe operation of the proposed facility.

The NRC staff finds that LME has appropriately identified and assessed the potential fire hazard and determined that the risk is very low and, therefore, has satisfied the requirements of 10 CFR 100.20(b) with respect to that hazard. In addition, the NRC staff finds that LME evaluated the potential fire hazard associated with nearby facilities and transportation routes and showed that the hazard does not pose undue risk to the proposed facility and, therefore, has satisfied the requirements of 10 CFR 100.21(e) with respect to that hazard. Additionally, as LME has conducted a safety assessment of the proposed site and facility and showed that the potential for accidents that could result in the release of significant quantities of radioactive fission products is extremely low with respect to a potential fire, the NRC staff finds that the requirements of 10 CFR 50.34(a)(1)(ii) have been satisfied with respect to that hazard.

2.2.6 Toxic Materials

2.2.6.1 Technical Evaluation

LME states in PSAR section 2.2.3.5 that in the proposed facility, actions by the operators of the control room are not needed to fulfill any required safety functions for safe operation of the reactors. Therefore, release of toxic gases causing harmful health effects on the control room operators was not assessed and discussed in the PSAR. The NRC staff notes that while there are no required safety functions (RSF) performed by operators, PSAR sections 5.5.3 and 7.3.12 describe one NSRST operator action required during some steam generator tube leak (SGTL) events to identify the leaking bundle, isolate it, and restore active cooling through the startup-shutdown system with the intact bundle. This is a safety-significant function that might be determined to be risk-significant once operator actions are assessed through the performance of additional risk-significance evaluations described in the response to RCI-1 dated August 14, 2026 (ML26230A155).

With the only safety-significant operator action being in response to a SGTL event, a preliminary assessment would be necessary if an initiating event were identified that could lead to both the release of toxic material and a SGTL. The only events within the scope of the licensing basis events identified by NRC staff that could lead to these failures occurring concurrently are external hazards such as seismic, flooding, or tornado hazards. PSAR section 6.4.12 describes how these events are addressed for the steam generator system (SGS) under the assessment

of PDC 2. The SGS is protected from flooding debris and wave runup as well as high winds, tornado, and tornado missiles by the RB and shield structure. With respect to seismic design, the SGS tubing is categorized as SC-IIA and designed to withstand the seismic event without tube leak or rupture. This is evaluated further in sections 3.7, 6.1, and 6.4.12 of this SE. Based on this information, the NRC staff finds that there are no initiating events within the scope of the licensing basis events that would lead to both the release of toxic material and a SGTL.

2.2.6.2 Conclusion

Based on the above technical evaluation, the NRC staff finds LME not assessing the release of toxic gases causing harmful health effects on the control room operators acceptable for the CP.

2.2.7 Conclusion

The NRC staff reviewed the discussion on potential hazards posed by the nearby facilities and transportation routes in PSAR section 2.2.2 and 2.2.3. The NRC staff used the guidance in DANU-ISG-2022-02, NUREG-0800 sections 2.2.1, 2.2.2, and 2.2.3, and relevant regulatory guides to evaluate the results presented in the PSAR. The NRC staff finds that LME has adequately identified the potential hazards to the proposed facility from nearby facilities and transportation routes using acceptable methods. Further, the NRC staff finds that LME has adequately evaluated the nature and proximity of man-related hazards to establish site characteristics for use in determining whether the LMGS design can accommodate commonly occurring hazards, and whether the risk of other hazards is very low and therefore has satisfied the requirements in 10 CFR 100.20(b). In addition, the NRC staff finds that LME evaluated potential hazards associated with nearby transportation routes, industrial and military facilities and established site characteristics such that potential hazards from those routes and facilities will pose no undue risk to the LMGS facility, as required in 10 CFR 100.21(e). The NRC staff also finds that LME has provided an adequate description and safety assessment of the site and a safety assessment of the facility, as required in 10 CFR 50.34(a)(1)(ii).

Based on its review, the NRC staff finds that LME has shown that hazards from these nearby facilities and transportation routes are not expected to develop into a design-basis hazard, except for crashes of a general aviation aircraft. LME has committed to conduct additional risk-informed assessments to be submitted with any OL application demonstrating either that hazards posed by a crash of a general aviation aircraft are below the hazard threshold or that the final design of the LMGS facility adequately addresses the hazard.

2.3 Meteorology

PSAR section 2.3 describes the meteorological phenomena used to evaluate the impacts of regional and local meteorology, on-site meteorological measurements, and short- and long-term dispersion estimates of normal and accidental releases for the LMGS. The section is divided into five subsections:

- Section 2.3.1, Regional Climatology
- Section 2.3.2, Local Meteorology
- Section 2.3.3, On-site Meteorological Measurements Program
- Section 2.3.4, Short-Term Dispersion Estimates for Accident Releases
- Section 2.3.5, Long-Term Dispersion Estimates for Normal Releases

The applicable regulatory requirements for CP applicants are as follows:

- 10 CFR 50.34(a)(1)(ii);
- 10 CFR 100.20(c)(2); and
- 10 CFR 100.21(c) and (d).

The applicable guidance for the evaluation of regional climatology, local meteorology, and atmospheric dispersion is as follows:

- DANU-ISG-2022-02, “Chapter 2, 'Site Information,'” March 2024 (ML23277A140);
- RG 1.76, “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants” (ML070360253);
- RG 1.221, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants” (ML110940300);
- RG 1.23, Revision 1, “Meteorological Monitoring Programs for Nuclear Power Plants.” (ML070350028);
- RG 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessment at Nuclear Power Plants.” (ML003740205);
- NUREG/CR-2858, “PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Plants.” (ML12045A149);
- RG 1.109, “Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I.” (ML003740384);
- RG 1.111, “Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors.” (ML003740354);
- NUREG/CR-2919, “XOQDOQ: Computer Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations.” (ML081360412); and
- RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants.” (ML031530505).

2.3.1 Regional Climatology

In accordance with 10 CFR 100.20(c)(2) and 10 CFR 100.21(d), and as discussed in DANU-ISG-2022-02, section 2.4, the application should describe meteorological characteristics at the site and the surrounding area, including sources for the severity of meteorological hazards used to establish the design bases, as reflected in the site characteristics. The guidance states the application should describe the overall climate of the region, including general airflow patterns

(wind direction and speed), temperature and humidity, precipitation (rain, snow, sleet, and freezing rain), potential influences from regional topography, and relationships between synoptic-scale atmospheric processes and local (site) meteorological conditions.

2.3.1.1 Technical Evaluation

The NRC staff reviewed PSAR section 2.3.1, which describes climate averages and extremes in the region of the LMGS site and regional meteorological phenomena, including a description of the general climate, frequencies of severe weather phenomena and annual freezing rain (ice storm) data, identification of National Centers for Environmental Information (NCEI) state climatic division(s), the LMGS site's air quality, and meteorological conditions including snowpack data, tornado parameters, wind data, and temperature data.

The regional climate data was obtained from the NCEI (formerly the National Climatic Data Center) Storm Events Database using data collected from 1950 to 2023 to extract statistics on weather in counties surrounding the LMGS site. The counties and zones included in the database search are listed in PSAR table 2.3-1. Other data sources used include National Oceanic and Atmospheric Administration (NOAA) 2022; ASHRAE, 2017; ASCE, 2016; USGCRP, 2017; and EIA, 2021. The applicant shows the National Weather Service (NWS) Stations surrounding the LMGS site in PSAR figure 2.3-2 and states that the Victoria NWS Station (KVCT) is approximately 25 mi (40 km) from the LMGS site and is the closest full-service weather station.

The NRC staff reviewed the information on the regional climate of the proposed LMGS site, and, confirmed that the applicant obtained the information from appropriate sources, and that the information appears reasonable for the geographic area. The NRC staff's assessments of the individual subsections are below.

General Climate

The NRC staff reviewed PSAR section 2.3.1.1, which includes an aerial map of the LMGS site and surrounding area as shown in PSAR figure 2.3-1. The applicant states that the LMGS site is located in a humid subtropical region and has summers with air temperatures reaching 90°F (32.2°C) or higher for about 100 days each year. The applicant states that thunderstorm activity occurs mainly in the spring and summer months and that the region is subject to occasional tropical storms and hurricanes that bring destructive winds and torrential rain.

The NRC staff reviewed the information on the general climate in the region of the proposed LMGS site, and based on its review, including confirmation that the applicant obtained the information from appropriate sources, and that the information appears reasonable for the geographic area, the NRC staff finds that the applicant's assessment is sufficient for evaluation of the general climate of the LMGS site. Therefore, the NRC staff finds that applicant's assessment of the general climate is acceptable.

Thunderstorms, Hail, and Lightning

The NRC staff reviewed PSAR section 2.3.1.2.1, which states that the applicant used data from NOAA to provide a summary of thunderstorm days by month recorded at the Victoria NWS Station (KVCT) as shown in PSAR table 2.3-2. KVCT is located approximately 24 mi (38 km) northeast of LMGS. The period of record for the meteorological station comprises 63 years. The

applicant notes that thunderstorms and associated lightning occur on an average of 52.6 days per year and thunderstorms occur most frequently from May through September in the LMGS site region with a peak of thunderstorms occurring in August.

The applicant used the NCEI Storm Events Database to extract statistics of observed hail near the LMGS site. A total of 652 hailstorm occurrences were documented during the summarized time period in the surrounding counties of Aransas, Bee, Calhoun, Colorado, DeWitt, Jackson, Knox, Lavaca, Matagorda, Refugio, Victoria, and Wharton, and approximately 38 percent of the hailstorm occurrences did not produce hailstones equal to, or larger than, 1 in. (2.54 cm) in diameter. The applicant states that from 1955 to 2023, 407 hail events produced hailstones documented as equal to, or larger than, 1 in. (2.54 cm) in diameter. The largest hailstones recorded during the referenced period were 4.5 in. (11.4 cm) on February 19, 1991, in DeWitt County; April 11, 1995, in Calhoun County; and June 20, 1996, in Wharton County, Texas.

The applicant states that the mean frequency of lightning strikes to earth is estimated using the method from the Electric Power Research Institute (EPRI) as recommended by the U.S. Department of Agriculture (USDA). This method assumes a correlation between the average number of thunderstorms per year and the number of lightning strikes to earth per square mile (mi^2) per year. The applicant states that the number of lightning strikes to earth per mi^2 per year is computed for the LMGS site based on the average number of thunderstorm days per year using the formula provided in PSAR section 2.3.1.2.1. The applicant calculated the frequency of lightning strikes to earth per mi^2 per year as 16.31 (6.30 strikes per km^2 per year) for the LMGS site.

The NRC staff reviewed the information on thunderstorms, hail, and lightning in the region of the proposed LMGS site, and based on its confirmation that the applicant obtained the information from appropriate sources, and that the information appears reasonable for the geographic area, the NRC staff finds that the applicant's assessment is sufficient for evaluation of potential impacts of thunderstorms, hail, and lightning at the site, to inform design bases for the facility. Therefore, the NRC staff finds that the applicant's assessment of thunderstorms, hail, and lightning is acceptable.

Tornadoes

The NRC staff reviewed PSAR section 2.3.1.2.2, which states that tornado intensity is classified according to the Enhanced Fujita Tornado Intensity Scale (EF-Scale) which the applicant outlines in PSAR table 2.3-3. The EF-Scale links wind speeds from a tornado event to the magnitude of observed damage according to 28 indicators. The EF-Scale ranks tornadoes in categories EF-1 to EF-5 using more specific structural damage indicators than the original and previously used tornado intensity classification scale, the Fujita Scale (F-Scale). The use of the EF-Scale is consistent with NRC guidance and is therefore acceptable.

The applicant used the NCEI Storm Events Database to extract statistics on regional tornadoes occurring in the counties in the region of the site. Based on the Storm Events Database search a total of 30 tornadoes have been reported in Calhoun County, Texas, where the LMGS site is located, from 1955 through 2018, with the highest rated tornadoes rated at F2 on the F-Scale. These occurred November 2, 1961; September 20, 1967; August 3, 1970; May 7, 1972; and October 29, 1972. Tornado path lengths spanned from less than 1 mi to 15.4 mi (1.6 km to 24.8 km) and tornado widths spanned from less than 1 ft to 200 ft (0.3 m to 70 m) in Calhoun

County. The applicant summarized the occurrences of tornadoes in PSAR table 2.3-4. The NRC staff recognizes NCEI as an appropriate source for information regarding tornado occurrences.

Regulatory Guide (RG) 1.76, Revision 1, specifies design-basis tornado characteristics by region in the United States. The applicant notes that the LMGS site is located in Region II as defined in RG 1.76. The applicable design-basis tornado maximum wind speed is 89 m/s. This wind speed consists of a translational speed of 18 m/s, and a maximum rotational speed of 72 m/s. The NRC staff considers RG 1.76 acceptable for use in selecting the design-basis tornado wind speeds and tornado-generated missiles that nuclear power plants should be designed to withstand.

PSAR section 3.7.4.10 discusses tornado wind loads and states that NSRST structures, including the shield structure (SST), are designed for tornadic forces and missiles. The applicant states that the tornado wind loads are computed in accordance with RG 1.76. There are three components to tornadic loading: (1) wind speed, (2) differential pressure, and (3) tornado-generated missile impact effect. The design wind speed is 200 mph (322 kph). Tornadoes cause a maximum pressure drop of 1.2 psi (8.3 kPa), dropping at a rate of 0.5 psi (3.4 kPa) per second. The applicant states that impact effects will be based on the design missile spectrum from RG 1.76, Revision 1, table 2. The applicant states that the enveloping values of Region II from RG 1.76 are used. Velocities listed are in the horizontal direction and the vertical velocity is taken as 67 percent of horizontal velocity. The automobile is considered to impact heights of 30 ft (9.1 m) or less from grade. The applicant notes that as described in RG 1.76, the NRC considers an automobile to be an appropriate missile in the evaluation of a plant's resistance to tornado damage.

The NRC staff reviewed the information on tornadoes in the region of the proposed LMGS site, and based on its review, including confirmation that the applicant obtained the information from appropriate sources, and that the information appears reasonable for the geographic area, the NRC staff finds that the applicant’s assessment is sufficient for evaluation of potential impacts from tornadoes at the site, and to inform design bases for the facility. Therefore, the NRC staff finds that the applicant’s assessment of tornadoes is acceptable.

Hurricanes and Tropical Storms

The NRC staff reviewed PSAR section 2.3.1.2.3, which states that the LMGS site is exposed to hurricanes and tropical storms. The applicant used the NCEI Storm Events Database to extract statistics from Calhoun County and surrounding counties of the LMGS site for the time period of 1950–2023. The applicant states that a total of 13 hurricanes recorded in the Storm Events Database have affected the area between 2003 and 2021, and from 1998 to 2021 58 tropical storms are reported in the Storm Events Database in Calhoun County and surrounding counties.

PSAR section 3.7.4.11 discusses hurricane wind loads and states that NSRST structures, including the SST, are designed for hurricane forces and missiles. The applicant states that the hurricane wind loads are computed in accordance with RG 1.221, which is guidance that the NRC staff considers acceptable for use in selecting the design-basis hurricane windspeed and hurricane-generated missiles that nuclear power plants should be designed to withstand. There are two components to hurricane loading, wind speed and hurricane-generated missile impact. The design wind speed is 210 mph (338 kph) taken from figure 1 of RG 1.221. The applicant based the impact effects on the design missile spectrum from tables 1 and 2 of RG 1.221 The

velocities listed in the table are in the horizontal direction and the vertical velocity is taken as 58 mph (93 kph) for all missiles. The applicant notes that the automobile need only be considered to impact heights of 30 ft (9.1 m) or less from grade.

The NRC staff reviewed the information on hurricanes and tropical storms in the region of the proposed LMGS site, and based on its review, including confirmation that the applicant obtained the information from appropriate sources, and that the information appears reasonable for the geographic area, the NRC staff finds that the applicant's assessment is sufficient for evaluation of potential impacts from hurricanes and tropical storms at the site, and to inform design bases for the facility. Therefore, the NRC staff finds that the applicant's assessment of hurricanes and tropical storms is acceptable.

Winds

The NRC staff reviewed PSAR section 2.3.1.2.4, which states that American Society of Civil Engineers (ASCE)/SEI 7-16 provides maps of 100-year mean return-interval wind gusts. The applicant determined that the 100-year return-interval 3 second gust wind speed at 29 ft (8.8 m) above ground level for the LMGS site is 122 mph (54.54 m/s) for risk factors I through IV using the ASCE 7 Hazard Tool.

The NRC staff reviewed the information on winds in the region of the proposed LMGS site, and based on its review, including verification that the applicant obtained the information from appropriate sources, in this case, ASCE 7-16, which is an industry standard for Minimum Design Loads and Associated Criteria for Buildings and Other Structures, the NRC staff finds that the applicant's assessment is sufficient for evaluation of potential impacts from winds at the site, and to inform design bases for the facility. Therefore, the NRC staff finds that the applicant's assessment of winds is acceptable.

Snow and Ice Storms

The NRC staff reviewed PSAR section 2.3.1.2.5, which describes that the majority of ice and snowstorm events occur at the LMGS site during December, January, and February. PSAR table 2.3-5 summarizes normal and maximum monthly snowfall recorded at the Victoria NWS Station (KVCT). The applicant states that the normal annual snowfall is 0.1 in. (0.25 cm). The maximum monthly snowfall is 2.1 in. (5.3 cm), recorded in January 1985, which the applicant states is also the maximum snowfall recorded in 24 hours.

The applicant states that glaze is a coating of generally clear and smooth ice which is formed by super-cooled drizzle/rain on exposed objects. The applicant cites a study from Tattelman and Gringorten (1973) which estimated glaze and wind loads for eight regions (I–VIII) of the contiguous United States. The estimates were based on observed ice storms in which the maximum ice thickness of 2.5 cm or more and 5 cm or more occurred during a 50-year period. The state of Texas is in region VI of the study. In the study the number of ice storms in 50 years with an ice thickness greater than or equal to 2.5 cm for region VI is 10 and the regional probability of an ice storm in one year with an ice thickness greater than or equal to 2.5 cm is 0.18. The applicant states that South Texas was not included in Tattelman and Gringorten's analysis due to the rarity of icing events, so the use of same data from Region VI for North/Central Texas is conservative.

The NRC staff reviewed the information on snow and ice storms in the region of the proposed LMGS site, and based on its review, including confirmation that the applicant obtained the information from appropriate sources, and that the information appears reasonable for the geographic area, the NRC staff finds that the applicant's assessment is sufficient for evaluation of potential impacts from snow and ice storms at the site, and to inform design bases for the facility. Therefore, the NRC staff finds that the applicant's assessment of snow and ice storms is acceptable.

Ambient Air Temperature

Summary from Regional Climatic Stations

The NRC staff reviewed PSAR section 2.3.1.2.6.1, which presents a summary of the temperature data in the area of the LMGS site. The applicant states that PSAR table 2.3-6 summarizes normal maximum and minimum temperatures recorded at the Victoria NWS Station (KVCT) between 1956 and 2022. Maximum temperatures usually occur between April and October, while minimum temperatures usually occur from November through March. The applicant states that the highest daily maximum temperature recorded at KVCT is 111°F (43.9°C) and lowest daily minimum temperature recorded is 9°F (-12.8°C).

The NRC staff reviewed the summary of the temperature data in the region of the proposed LMGS site, and based on its review, including confirmation that the applicant obtained the information from appropriate sources, and that the information appears reasonable for the geographic area, the NRC staff finds that the applicant's assessment is sufficient for evaluation of potential impacts from temperature at the site, and to inform design bases for the facility. Therefore, the NRC staff finds that the applicant's summary of the temperature data is acceptable.

Design Dry Bulb and Mean Coincident Wet Bulb Temperatures

The NRC staff reviewed PSAR section 2.3.1.2.6.2, which states that the mean dry bulb temperature (DBT) at KVCT is 70.6°F (21.4°C) with a mean maximum dry bulb temperature of 84.7°F (29.3°C) and mean minimum dry bulb temperature of 54.0°F (12.2°C). The applicant states that the mean wet bulb temperature (WBT) at KVCT is 63.0°F (17.2°C) with a mean maximum wet bulb temperature of 75.2°F (24.0°C) and mean minimum wet bulb temperature of 47.7°F (8.7°C).

The applicant states that the design-basis DBTs and WBTs were estimated for the LMGS site using local meteorological data. The temperatures include:

- Normal DBT
- Mean DBT
- Mean WBT
- Maximum Normal DBT
- Maximum Mean DBT
- Maximum Mean WBT
- Minimum Normal DBT
- Minimum Mean DBT
- Minimum Mean WBT

- DBT with Annual Exceedance Probability of 0.4 percent
- Mean Coincident WBT With Annual Exceedance Probability of 0.4 percent
- DBT With Annual Exceedance Probability of 1.0 percent
- Mean Coincident WBT With Annual Exceedance Probability of 1.0 percent
- DBT With Annual Exceedance Probability of 2.0 percent
- Mean Coincident WBT With Annual Exceedance Probability of 2.0 percent
- WBT (Non-Coincident) With Annual Exceedance Probability of 0.4 percent
- WBT (Non-Coincident) With Annual Exceedance Probability of 1.0 percent

The applicant provides the bounding values for the design of dry bulbs and mean coincident wet bulb temperatures in PSAR table 2.3-7.

The NRC staff reviewed the information on the design-basis DBTs and WBTs for the proposed LMGS site, and based on its review, including confirmation that the applicant obtained the information from appropriate sources, and that the information appears reasonable for the geographic area, the NRC staff finds that the applicant's assessment is sufficient for evaluation of potential impacts from design-basis DBTs and WBTs at the site, and to inform design bases for the facility. Therefore, the NRC staff finds that the applicant's assessment of design-basis DBTs and WBTs is acceptable.

Fog

The NRC staff reviewed PSAR section 2.3.1.2.7, which discusses fog in the region for the LMGS site. Fog is defined as water droplets suspended in the atmosphere in the vicinity of earth's surface that affect visibility. Fog reduces visibility below 1 km. Heavy fog is defined in NUREG 1555, section 2.7 as fog with visibility less than 0.4 km. The applicant stated that the number of days with heavy fog at KVCT is summarized in the environmental report (ER) table 2.7-6. The applicant notes that heavy fog occurred an average of 46.9 days per year in the time period from 1964 to 2022, with the most occurrences in November, December, and January.

The NRC staff reviewed the information on fog in the region for the proposed LMGS site, and based on its review, including confirmation that the applicant obtained the information from appropriate sources, and that the information appears reasonable for the geographic area, the NRC staff finds that the applicant's assessment is sufficient for evaluation of potential impacts fog at the site, and to inform design bases for the facility. Therefore, the NRC staff finds that the applicant's assessment of fog is acceptable.

Precipitation

The NRC staff reviewed PSAR section 2.3.1.2.8, which states that the maximum monthly precipitation recorded at KVCT was 20.34 in. (51.7 cm) in July 2007 as shown in PSAR table 2.3-11, and that the maximum monthly snowfall recorded was 2.1 in. (5.3 cm) in January 1985 as shown in PSAR table 2.3-5.

The NRC staff reviewed the information on precipitation in the region for the proposed LMGS site, and based on its review, including confirmation that the applicant obtained the information from appropriate sources, and that the information appears reasonable for the geographic area, the NRC staff finds that the applicant's assessment is sufficient for evaluation of potential

impacts of precipitation at the site, and to inform design bases for the facility. Therefore, the NRC staff finds that the applicant's assessment of precipitation is acceptable.

2.3.1.2 Conclusion

The NRC staff reviewed the regional climatology information used to characterize the area of the LMGS site. The NRC staff's review confirmed that the applicant utilized appropriate sources for the area's climate information, and that the information is reasonably representative of the geographic area. Based on its review, the NRC staff determined that the applicant's characterization of the regional climatology is acceptable.

2.3.2 Local Meteorology

2.3.2.1 Technical Evaluation

The NRC staff reviewed PSAR section 2.3.2, in which the applicant states that for the purposes of identifying a climate region with meteorological stations relatively close to the LMGS site and subject to the same general climatic influences, the analysis concentrates on KVCT located in Victoria County, Texas, and the Palacios-R. B. Trull Municipal Airport (KPSX) located in Matagorda County, Texas. The applicant states that KVCT is located approximately 25 mi (40.2 km) to the northwest of the LMGS site and KPSX is roughly 35 mi (56.3 km) to the northeast. The applicant notes that nearby meteorological stations to the LMGS site are listed in PSAR table 2.3-8, that their locations are shown in PSAR figure 2.3-2, and that publicly available weather observations from KVCT and KPSX Stations were obtained from the NWS and NCEI.

The applicant states that to meet the intent of 10 CFR 100.20(c)(2) and the guidance contained in RG 1.23 "Meteorological Monitoring Programs for Nuclear Power Plants" (ML070350028), an applicant must show that meteorological data to be used in analyses that support a 10 CFR Part 50 CP application are representative of site conditions. The applicant notes that the regional airports listed in the previous paragraph are in the same climate zone along the western Gulf Coast and terrain is relatively flat and meteorological conditions are expected to be similar. The applicant states that wind roses showing the distribution of wind speed and directions at all four regional airports surrounding the LMGS site are similar, and a correlation analysis was performed using meteorological data from Rockport, Victoria, and Palacios that showed correlations with air temperature, wind speeds, and wind directions. The applicant states that the moderate and high correlations among the data sets support the use of airport data in applicable portions of PSAR section 2.3.2 and Part III of the CP application (ER) because both the LMGS site and KVCT fall within the geographic area among these three airports, the KVCT NWS Station has been determined to be an accurate source for meteorological data comparable to the LMGS site. The applicant notes that the LMGS site is located on the Texas Gulf Coast, northeast of San Antonio Bay, 60 mi (97 km) northeast of Corpus Christi and 128 mi (206 km) southeast of San Antonio and is at an elevation of 27 ft (8.23 m) above sea level and surrounded by agricultural land.

The NRC staff reviewed the information on the local meteorology of the proposed LMGS site and confirmed that the applicant obtained the information from appropriate sources, and that the information and data used appears reasonable for the geographic area. The NRC staff's assessments of the individual subsections are below.

2.3.2.1.1 Normal and Extreme Values of Meteorological Parameters

Wind and Air Flow

The NRC staff reviewed PSAR section 2.3.2.1.1, in which the applicant states that wind data will be collected from the meteorological (MET) tower that will be placed in service at the LMGS site and will be provided with the OL application.

The applicant states that the mean wind speed recorded at the KVCT NWS Station is 9 4 mph (15.1 kph). Wind data from the KVCT NWS Station spans 39 years from 1983 to 2022. The applicant notes that the maximum 2-min. wind speed recorded is 62 mph (99.8 kph) which occurred in July 2003 and the maximum 3-s wind speed was recorded in August 2017 at 83 mph (133.6 kph).

The applicant provides monthly wind roses for January 2013 through December 2023 in PSAR figures 2.3-3 through 2.3-14 for KVCT and PSAR figures 2.3-15 through 2.3-26 for Palacios-R. B. Trull Municipal Airport (KPSX).

The applicant states that the South Texas Project (STP) Nuclear Generating Station (NGS) tower is the nearest source with published, representative meteorological data that provides Pasquill stability class computed in accordance with RG 1.23, Revision 1, March 2007. The applicant states that summaries of hourly wind and Pasquill data from the STP NGS dated 2017 through 2021 were used for the data set to compute atmospheric dispersion factors (X/Q) and the joint frequency distribution (JFD) at the LMGS site. The joint data recovery (utilized for JFD development) for 2017 through 2021 from the STP NGS tower exceeded the 90 percent data completeness requirement for all years. Wind data obtained from the STP NGS tower used for the JFD and containing the wind speed, direction, and Pasquill stability class are provided in PSAR table 2.3-13 and were extracted from publicly available effluent reports. The applicant states that JFD data will be collected from the on-site MET tower that will be placed in service at the LMGS site and will be provided with the OL application.

The NRC staff reviewed the information on wind and air flow in the region for the proposed LMGS site, and based on its review, including confirmation that the applicant obtained the information from appropriate sources, and that the information appears reasonable for the geographic area, the NRC staff finds that the applicant's assessment is sufficient for evaluation of potential impacts of wind and air flow at the site, and to inform design bases for the facility. Therefore, the NRC staff finds that the applicant's assessment of wind and air flow is acceptable.

Air Temperature

The NRC staff reviewed PSAR section 2.3.2.1.2 in which the applicant states that temperature data for Victoria are presented in PSAR table 2.3-6. The applicant states that normal temperatures range from the mid to upper 50s ºF (approximately 14ºC) in the winter to the low to mid 80s ºF (approximately 29ºC) in the summer. Normal daily maximum temperatures range from about 65°F (18ºC) in mid-winter to about 95°F (35ºC) in mid-summer. The applicant states that the normal daily minimum temperatures range from about 44°F (6.7ºC) in mid-winter to about 74°F (23ºC) in mid-summer and the extreme daily maximum recorded was 111°F (41ºC) (September 2000), while the extreme daily minimum was 9°F (-13ºC) (December 1989).

The NRC staff reviewed the information on air temperature in the region for the proposed LMGS site, and based on its review, including confirmation that the applicant obtained the information from appropriate sources, and that the information appears reasonable for the geographic area, the NRC staff finds that the applicant's assessment is sufficient for evaluation of potential impacts of air temperature at the site, and to inform design bases for the facility. Therefore, the NRC staff finds that the applicant's assessment of air temperature is acceptable.

Atmospheric Moisture

The NRC staff reviewed PSAR section 2.3.2.1.3, in which the applicant states that long-term relative humidity and absolute humidity data for the Victoria NWS Station (KVCT) are presented in PSAR table 2.3-9. Relative humidity values are consistent throughout the year and average in the mid to upper 70 percent range. The mean absolute humidity is greatest in the summer months.

The NRC staff reviewed the information on atmospheric moisture in the region for the proposed LMGS site, and based on its review, including confirmation that the applicant obtained the information from appropriate sources, and that the information appears reasonable for the geographic area, the NRC staff finds that the applicant's assessment is sufficient for evaluation of potential impacts of atmospheric moisture at the site, and to inform design bases for the facility. Therefore, the NRC staff finds that the applicant's assessment of atmospheric moisture is acceptable.

Precipitation

Rain

The NRC staff reviewed PSAR section 2.3.2.1.4, in which the applicant states that on-site precipitation observations are not available from the LMGS site, so hourly data collected at KVCT (approximately 25 mi [40 km] northwest of the LMGS site) is being used as an alternative because it is the nearest data source to the LMGS site. The location of KVCT is shown in PSAR figure 2.3-2.

The applicant notes that the monthly normal precipitation data recorded at KVCT are presented in PSAR table 2.3-10. Precipitation falls an average of about 95 days per year, and the normal annual precipitation is about 40.4 in. (102.6 cm). The applicant states that the maximum monthly rainfall ranges from about 7 in. (17.8 cm) to just over 20 in. (50.8 cm). The minimum monthly amount was trace (less than the lowest measurable amount) to 0.01 in. (0.025 cm) in 1998. The maximum in 24 hours recorded was 9.87 in. (25.1 cm) in April 1991. May and September are normally the wettest months of the year. Maximum and minimum precipitation for KVCT is provided in PSAR table 2.3-11 for the period of record from 1960 to 2022 (NOAA, 2022).

Snow

The applicant states that appreciable snowfall is relatively infrequent in the area. The applicant states that snowfall data are summarized in PSAR table 2.3-5 for KVCT (NOAA, 2022) and discussed in PSAR section 2.3.1.2.

The NRC staff reviewed the information on precipitation in the region for the proposed LMGS site, and based on its review, including confirmation that the applicant obtained the information from appropriate sources, and that the information appears reasonable for the geographic area, the NRC staff finds that the applicant's assessment is sufficient for evaluation of potential impacts of precipitation at the site, and to inform design bases for the facility. Therefore, the NRC staff finds that the applicant's assessment of precipitation is acceptable.

Fog

The NRC staff reviewed PSAR section 2.3.2.1.5, in which the applicant states that heavy fog is defined as the occurrence of fog reducing the horizontal visibility to 0.25 mi (0.40 km) or less. Fog data for KVCT is presented in PSAR table 2.3-12. The applicant states that at KVCT heavy fog was observed on an average of approximately 47 days per year and occurred most often during November through January, with the peak months having up to seven days per month. July and August experience the fewest days of heavy fog and average less than one day per month.

The NRC staff reviewed the information on fog in the region for the proposed LMGS site, and based on its review, including confirmation that the applicant obtained the information from appropriate sources, and that the information appears reasonable for the geographic area, the NRC staff finds that the applicant's assessment is sufficient for evaluation of potential impacts fog at the site, and to inform design bases for the facility. Therefore, the NRC staff finds that the applicant's assessment of fog is acceptable.

Atmospheric Stability

The NRC staff reviewed PSAR section 2.3.2.1.6, in which the applicant states that the general climate of the LMGS site is maritime subtropical (or humid subtropical), which is characterized by mild, short winters, long periods of mild sunny weather in autumn, windy but mild weather in spring, and long hot summers. The applicant notes that maritime tropical air mass characteristics prevail much of the year, especially during the summer with the establishment of the Bermuda High and the Gulf of America (formerly known as Gulf of Mexico) High. The applicant states that in the central Gulf Coast states, air stagnation conditions exist 5–10 percent of the time, and in July and August, from southeastern New Mexico to southern Texas there are more than 10 days per month of air stagnation with a maximum of about 15 days close to the Mexican border. After August the number of air stagnation occurrences per month decreases quickly.

The applicant states that the land/sea temperature contrast during summer days creates a circulation forming a sea breeze where cooler, more saturated air pushes inland as the warm air rises inland, but the opposite occurs at night, where inland plains cool rapidly while the sea stays relatively warmer, thus causing a breeze to push offshore into the Gulf of America.

Inversion Persistence

The applicant states that during the day, dry air naturally cools with increased altitude, and solar radiation warms the earth's surface, and convection creates winds and gusts that transport air vertically. In the evening when the sun sets the earth's surface is not heated and, as a result, the ground and the air adjacent to the ground cool more rapidly than the overlying atmosphere. The applicant notes that heat from the warmer air is transferred back to earth, creating a layer of

cooler, denser air near the surface which creates a temperature inversion, where cool air at surface level has warmer air above it through the lowest levels of the atmosphere.

Mixing Heights

The applicant states that mixing heights for Victoria, Texas, were obtained from the U.S. Environmental Protection Agency (USEPA) Support Center for Regulatory Atmospheric Modeling for the year 1991. The applicant states that the USEPA data is more current than the traditional Holzworth reference that is often used. The applicant states that mixing heights vary during the day, with the lowest average mixing heights (approximately 344 m) generally recorded during the morning hours in the winter and the highest readings (approximately 1414 m) recorded during the afternoon in the summer. The applicant lists the approximate seasonal and annual estimates of mixing heights for the LMGS site as follows:

- Winter (December, January, February): 344 m (morning), 666 m (afternoon).
- Spring (March, April, May): 466 m (morning), 943 m (afternoon).
- Summer (June, July, August): 566 m (morning), 1414 m (afternoon).
- Autumn (September, October, November): 346 m (morning), 1289 m (afternoon).
- Annual: 433 m (morning), 1077 m (afternoon).

The NRC staff reviewed the information on atmospheric stability in the region for the proposed LMGS site, and based on its review, including confirmation that the applicant obtained the information from appropriate sources, and that the information appears reasonable for the geographic area, the NRC staff finds that the applicant's assessment is sufficient for evaluation of potential impacts of atmospheric stability at the site, and to inform design bases for the facility. Therefore, the NRC staff finds that the applicant's assessment of atmospheric stability is acceptable.

2.3.2.1.2 Potential Influence of the Plant and Its Facilities on Meteorology

The NRC staff reviewed PSAR section 2.3.2.2, in which the applicant states that plant systems that may have noticeable effects on the local meteorology are pollutant sources around the LMGS site. These sources include stacks from standby generators. LME also states that there will be minor impacts on local air quality during construction. The applicant notes that the LMGS site uses dry cooling towers (air cooled condensers) instead of mechanical draft cooling towers; therefore, there is no impact from the cooling system on the local environment from plume shadowing, fogging, or icing.

The NRC staff reviewed the information regarding the potential influence of the plant and its facilities on meteorology for the proposed LMGS site and based on its review of the information provided, the NRC staff finds that the applicant's assessment is sufficient for evaluation of the potential influence of the plant and its facilities on meteorology. Therefore, the NRC staff finds that the applicant's assessment is acceptable.

2.3.2.1.3 Local Meteorological Conditions for Design and Operation

The applicant states that the local meteorological conditions for the design and operational basis are provided in the design-basis discussions in PSAR section 2.3.1. The NRC staff evaluated the information from PSAR section 2.3.1 in the regional climatology section of this SE.

2.3.2.2 Conclusion

The NRC staff reviewed the information on local meteorology in the region for the proposed LMGS site, and based on its review, including confirmation that the applicant obtained the information from appropriate sources, and that the information appears reasonable for the geographic area, the NRC staff finds that the applicant's assessment is sufficient for evaluation of local meteorology in the region of the site, and to inform design bases for the facility. Therefore, the NRC staff finds that the applicant's assessment of local meteorology in the region is acceptable.

2.3.3 On-Site Meteorological Measurements Program

The NRC staff reviewed PSAR section 2.3.3, in which the applicant notes that for the LMGS CP application, meteorological data are obtained from local meteorological stations and the STP NGS meteorological tower (see PSAR table 2.3-8 and PSAR figure 2.3-2). The applicant states that a meteorological tower is planned for installation at the LMGS and for data collection during plant operation. The applicant states that during construction of the LMGS the primary source of meteorological data is KVCT. LME plans to build a meteorological tower that meets the requirements of RG 1.23, Revision 1, to be the primary source of meteorological data for operations of LMGS. The applicant states that post-application data collection begins as soon as the meteorological tower is operational. PSAR section 2.3.3.1 describes the on-site meteorological tower to be placed in operation at the LMGS.

2.3.3.1 Technical Evaluation

2.3.3.1.1 Meteorological Instrumentation

The NRC staff reviewed PSAR section 2.3.3.1, in which the applicant states that the primary source of meteorological data during operations is an on-site multi-level tower, and that the guidance of RG 1.23 for a basic on-site meteorological monitoring system will be met. The applicant states that details of the on-site tower include:

- Wind speed and direction is measured on one open-lattice tower or mast at approximately 10 m and 60 m above ground level (AGL).
- Vertical temperature difference is measured on the same open-lattice tower or mast as wind speed and wind direction between the 10 m and 60 m AGL levels.
- Ambient temperature is monitored at approximately 10 m AGL.
- Precipitation is measured near ground level near the base of the tower or mast.

- Atmospheric moisture is not planned to be monitored because the site will be utilizing air cooled condensers, which does not require monitoring atmospheric moisture as described in RG 1.23.

- The siting of meteorological instruments meets the guidance of RG 1.23, Revision 1, Regulatory Position C.3.

- The time-average accuracies for digital systems meets the criteria listed in table 2 of RG 1.23, Revision 1, and the ambient temperature instrumentation is capable of operating over the range of expected climatic extremes based on regional climatology.

- Instrument maintenance and servicing schedules follows the guidance of RG 1.23, Revision 1, Regulatory Position C.5.

- The data reduction and compilation guidance of RG 1.23, Revision 1, Regulatory Position C.6, is followed.

- Per RG 1.23, Revision 1, Regulatory Position C.8, the meteorological monitoring system is able to support emergency preparedness.

The NRC staff reviewed the information regarding meteorological instrumentation for the proposed LMGS site, and based on its review, including confirmation that the applicant will use relevant NRC guidance for the on-site meteorological monitoring system and that the information presented in PSAR section 2.3.3.1 is consistent with that of RG 1.23, the NRC staff finds that the applicant's assessment is sufficient for evaluation of the proposed meteorological instrumentation at the site. Therefore, the NRC staff finds that the applicant's assessment of the proposed meteorological instrumentation is acceptable.

2.3.3.1.2 Plans to Access Local Meteorological Data during License Period

The NRC staff reviewed PSAR section 2.3.3.2, in which the applicant states that meteorological data to support development of the CP application is collected from nearby airports. These airports include KVCT and the Palacios-R. B. Trull Municipal Airport (KPSX). PSAR section 2.3.2 provides climatic data from the airports.

The applicant states that meteorological instruments installed at airports meet FAA requirements as described in the FAA Automated Weather Observing Systems for Non-Federal Applications Advisory Circular. The applicant lists the FAA specifications for meteorological instruments in PSAR table 2.3.3-1.

The applicant notes that the FAA performs calibration and maintenance of meteorological systems at airports, and on-site instrument calibration is required annually unless more frequent calibration is specified by the FAA region. The applicant states that calibrations are required to be performed by a qualified technician with FAA verification authority and witnessed by a qualified FAA non-Federal inspector, and that Facilities Maintenance Log and Technical Performance Record forms are maintained. Meteorological data from airports are archived by the NCEI. The applicant states that NCEI subjects' surface meteorological data to quality assurance and analysis procedures described in Del Greco et al., 2006.

The NRC staff reviewed the information regarding plans to access local meteorological data for the proposed LMGS site until the onsite data can be collected when the onsite meteorological tower is completed. Based on its review, including confirmation that the applicant is obtaining the information from appropriate sources that use quality assurance procedures, and that the plan appears reasonable for the geographic area, the NRC staff finds that the applicant's plan to access local meteorological data is sufficient and, therefore, acceptable for the LMGS site.

2.3.3.1.3 Plans to Access Local Meteorological Data During Emergencies

The NRC staff reviewed PSAR section 2.3.3.3, in which the applicant states that meteorological measurements from the on-site tower and local airports are available for use in responding to accidental radiological releases or other emergencies and for other routine purposes during the licensing period.

The applicant states that when needed during an emergency, real-time hourly surface meteorological measurements of wind direction, wind speed, air temperature, cloud amount/height, and weather type observed at the airports are available from the NWS. The applicant plans to use the hourly surface weather observations recorded at KVCT, Victoria, Texas. The applicant states that if weather observations are not available from KVCT, then weather information from another station with hourly meteorological data in the Site Climate Region is used. The applicant states the following stations may be accessed:

- Palacios-R.B. Trull Municipal Airport (KPSX).
- Calhoun County-Port Lavaca Airport (KPKV).
- Aransas County Airport Rockport (KRKP).

The NRC staff reviewed the information regarding plans to access local meteorological data during emergencies for the proposed LMGS site. Based on the fact that data from the on-site tower and data from local airports will be available, and based on its review, including confirmation that the applicant obtained the information from appropriate sources, and that the information regarding the nearby airports and the meteorological measurements available is consistent with the geographic area, the NRC staff finds that the applicant's assessment of its plans to access local meteorological data is sufficient and therefore, acceptable for the LMGS site.

2.3.3.2 Conclusion

The NRC staff reviewed details of the on-site meteorological measurements program provided by the applicant in the PSAR. Based on the NRC staff's review as outlined in the sections above, the NRC staff finds the applicant's plan for on-site meteorological measurements program acceptable.

2.3.4 Short-term Dispersion Estimates for Accident Releases

2.3.4.1 Technical Evaluation

The NRC staff reviewed PSAR section 2.3.4.1, in which the applicant states that estimates of the local atmospheric dispersion factors (X/Qs) for LMGS are developed using an atmospheric

dispersion model and conservative assumptions. The applicant states that a bounding, non-directional, non-location-specific methodology is used to calculate the X/Qs to the control room, EAB, and LPZ for all licensing basis events. This methodology is used in PSAR section 2.3.4.2 and PSAR section 2.3.4.3. The applicant states that for LMGS, the EAB and the LPZ boundary are identical. PSAR section 3.1.5 summarizes the applicant's atmospheric dispersion and dose calculation methodology.

2.3.4.1.1 Control Room Atmospheric Dispersion Factor

The NRC staff reviewed PSAR section 2.3.4.2, in which the applicant states that the control room calculation methodology is based on the diffuse point source - point receptor methodology described in RG 1.194. The applicant states that the atmospheric dispersion Equation 8 of RG 1.194 is appropriate for use because it applies when the assumed activity is released from many sources on the surface of the building and the difference in elevation between the source and receptor is greater than 30 percent of the height of the upwind building. This applies to LMGS with the RB or the fuel handing auxiliary building (FHAB) as the source and the controls and electrical building (CEB) containing the control room as the receptor.

The applicant states that the following assumptions are used:

- The radiological release for calculating control room dose is assumed to originate from the FHAB. Anticipated operational occurrences (AOOs) and design-basis events (DBEs) can result in potential releases from depressurization or from system leakage. For depressurizations, only releases from the RB and FHAB occur within the AOO or DBE range. Thus, full depressurizations into the HSF are only possible as a design-basis accident (DBA) or a beyond-design-basis event because the reliability of isolation of those systems keeps these events out of the AOO or DBE range. For system leakage, the fuel handling system (FHS) isolates at high temperature to protect the system from transients that could damage FHS equipment. This is consistent with the helium services system, which isolates should leakage occur. As such, system leakage during AOOs and DBEs can only come from within the RB or FHAB.

  Since the FHAB is closer to the CEB, which houses the control room, than the RB, the FHAB is used as the radiological release location.

- Distance from the edge of the FHAB to the edge of the CEB is 59 m. (This distance conservatively neglects the distance from the nearest point of the CEB to the CEB intake). Additionally, 50 m is used for further conservatism.

- No credit is taken for differences in building height between the point source (RB) and point receptor (CEB).

The applicant states that X/Q used for the control room is 6.66E-03 seconds per cubic meter (s/m^3).

The NRC staff reviewed the methodology and assumptions used by the applicant in its control room atmospheric dispersion analysis. The NRC staff finds the calculation of these generic dispersion values to be consistent with the guidance of RG 1.194 and the methodology outlined

in the topical report (TR) and, therefore, acceptable used for analysis in this CP application. LME states in the PSAR that for the OL application, they will perform a dispersion analysis with site-specific data to either affirm that the CP application analysis results using these generic dispersion factors are conservatively bounding or provide site-specific analysis results with dispersion factors calculated with the data from the onsite meteorological tower.

2.3.4.1.2 Exclusion Area Boundary and Low Population Zone Atmospheric Dispersion Factors

The NRC staff reviewed PSAR section 2.3.4.3 in which the applicant states that EAB and LPZ X/Qs are calculated for DBAs and non-design-basis accidents (non-DBAs) based on the methodology contained in RG 1.145, Revision 1, with some adjustments. The applicant states that the LMGS EAB and LPZ boundary have both been established at 400 m and therefore, a single set of X/Q values have been developed that apply to both the EAB and LPZ. The applicant states that equations (1), (2), and (3) in RG 1.145 are used to calculate a bounding X/Q at the EAB/LPZ. The inputs and assumptions used in the calculation are listed in PSAR section 2.3.4.3. Using the methodology described in RG 1.145 and the assumptions listed in the PSAR, the calculated X/Qs at the EAB/LPZ are 3.57E-04 s/m^3 for DBAs and 1.89E-04 s/m^3 for non-DBEs.

The NRC staff reviewed the methodology and assumptions used by the applicant in its EAB and LPZ atmospheric dispersion analysis. PSAR table 1.4.4 lists the Xe-100 Licensing Topical Report for the Atmospheric Dispersion and Dose Calculation Methodology, Revision 2 (ML23268A456) (Atmospheric Dispersion and Dose Calculation TR) as being incorporated by reference in the PSAR and applicable to section 2.3.4. The EAB and LPZ methodology and dispersion values are the same as the methodology and dispersion values outlined in the Atmospheric Dispersion and Dose Calculation TR. The NRC staff finds the calculation of these generic dispersion values to be consistent with the guidance of RG 1.145 and the methodology outlined in the Atmospheric Dispersion and Dose Calculation TR and, therefore, acceptable to be used for analysis in this CP application. LME states in the PSAR that for the OL application, they will perform a dispersion analysis with site-specific data to either affirm that the CP dispersion analysis results using these generic dispersion factors are conservatively bounding or provide site-specific analysis results with dispersion factors calculated with the data from the onsite meteorological tower.

2.3.4.2 Conclusion

The NRC staff reviewed the methodology and assumptions used by the applicant in its control room, EAB, and LPZ atmospheric dispersion analyses. Based on the NRC staff's review outlined in the sections above, the NRC staff finds the dispersion analyses acceptable for use in this CP application.

2.3.5 Long-term Dispersion Estimates for Normal Releases

2.3.5.1 Technical Evaluation

The NRC staff reviewed PSAR section 2.3.5.1, in which the applicant states that the estimates of long-term atmospheric dispersion factors (X/Qs) and deposition factors (D/Qs) for LMGS are made using a straight-line Gaussian model, consistent with the requirements of RG 1.109 and RG 1.111. The applicant uses the NRC's XOQDOQ dispersion and deposition model as described in NUREG/CR-2919. The applicant's objective is to calculate X/Q and D/Q values for

releases originating at the LMGS NI release envelope out to a distance of 50 mi (80.5 km). The LMGS RB, FHAB, and HSF are all included within the perimeter of the release envelope.

The applicant states that the X/Q and D/Q values are determined at the site boundary and at offsite receptors. General X/Q factors for graduated distances from the release envelope out to 50 mi (80.5 km) are also calculated. Offsite receptors are identified in each of the sixteen directional sectors within 50 mi (80.5 km) of the LMGS center point. The applicant provides a complete list of the offsite receptors considered in PSAR table 2.3.5-1.

2.3.5.1.1 Calculations

The NRC staff reviewed the dispersion estimates the applicant made for LMGS. The applicant states that it uses an atmospheric dispersion model prepared in accordance with the guidance provided in RG 1.111 and uses meteorological data obtained from STP for the period from January 2017 through December 2021. The applicant states that STP data are used due to its proximity to LMGS (approximately 50 mi [80 km] east-northeast of LMGS), similarity in topography, and distance to the coast. The applicant states that the meteorological data in the form of JFDs are extracted from publicly available STP annual effluent reports from the NRC and contain JFDs of wind speed, direction, and Pasquill stability class (STP 2018; STP 2019; STP 2020; STP 2021; STP 2022).

In PSAR section 2.3.5.2, the applicant describes the following assumptions included in the analysis:

- Effluent Radioactive Decay
- Wet Deposition
- Maximum Wind Speed
- Release Level
- Building Wake Effect Parameter
- Calm Meteorological Data

The applicant states that X/Q (dispersion) and D/Q (deposition) factors are calculated at locations as a function of the downwind distance from LMGS. The locations include receptors on the LMGS site boundary in each of the 16 directional sectors and at 18 offsite locations. In addition, the applicant states that general X/Q and D/Q factors are calculated radially out to 50 mi (80.5 km) from the LMGS effluent release points. For all the locations considered, the X/Q factors are calculated considering scenarios of no plume depletion/no decay, no plume depletion / 2.26 day half-life, and plume depletion / 8 day half-life. The X/Q and D/Q factors are calculated as average values over a period of one year.

The applicant provides the results of the long-term X/Q and D/Q calculations in PSAR tables 2.3.5-2 through 2.3.5-7. The applicant states that based on the analyses, the established site characteristic value for the maximum average annual dispersion factor at the site boundary is a value of 1.00E-04 s/m^3 (southwest sector) for any given sector and the established site characteristic value for the maximum average annual deposition factor at the site boundary is a value of 3.50E-07 ($1/m^2$). These values are shown in PSAR table 2.3.5-2. The scope of the NRC staff's review was limited to assessing the proposed methodology used to perform the calculations since the preliminary analysis of the atmospheric dispersion factors is based on meteorological data from STP. Therefore, the NRC staff did not assess the preliminary

meteorological data used and have not approved such data for use in additional licensing actions without approval by the NRC staff. The applicant states in the PSAR that the OL application will include the atmospheric dispersion analysis using site-specific meteorological data collected from an onsite met tower constructed in accordance with RG 1.23, and that the applicant will compare that analysis to the PSAR analysis and then use the more conservative values. While the scope of the NRC review did not include a confirmatory analysis for the dispersion values, the NRC staff did review the methodology and assumptions used by the applicant in its long-term atmospheric dispersion analysis and finds the methodology and assumptions used to be consistent with RG 1.111.

2.3.5.2 Conclusion

The NRC staff finds the calculation of these long-term dispersion values to be consistent with the guidance of RG 1.111 and, therefore, acceptable to be used in the calculation of annual doses from routine releases with the understanding that, as the applicant states in the PSAR, it will perform a long-term dispersion analysis with site-specific data from LMGS at the OL stage to either: 1) affirm that dose calculations using the long-term dispersion values in the CP PSAR are conservatively bounding, or 2) provide site-specific analysis results using long-term dispersion factors developed during site characterization.

2.3.6 Conclusion

The NRC staff evaluated the proposed LMGS site meteorology, which included a description of the general climate of the region, local meteorological conditions, and atmospheric dispersion as described in PSAR section 2.3. The NRC staff confirmed that the information presented in the PSAR was obtained from credible sources. The NRC staff also reviewed how that information was used in assessing the general climate of the region and the meteorological conditions that are used in determining the design and operation of the facility. In addition, the NRC staff reviewed LMGS's atmospheric dispersion analysis for the site, which will be used in the radiation dose consequences analysis.

The NRC staff finds that the site characteristics associated with meteorology, including general and local climatology, are acceptable and reasonably representative of the proposed LMGS site region. The NRC staff finds that the data resources (e.g., meteorological data sources) used to prepare PSAR section 2.3 are appropriate. The NRC staff finds that LMGS considered credible meteorological events in developing the design-basis parameters for the proposed facility.

Based on its review, the NRC staff finds that the level of detail and analyses provided on LMGS site meteorology demonstrates an adequate design basis and satisfies the applicable acceptance criteria of DANU-ISG-2022-02, "Chapter 2, 'site information'", allowing the NRC staff to find that:

- The meteorological history and projections for the proposed site have been prepared in an acceptable form.

- These projections have been factored into the choice of facility location and design sufficiently to provide assurance that no weather-related event is likely to cause damage to the reactor facility during its lifetime that could result in the uncontrolled release of radioactive material to the unrestricted area.

The NRC staff finds the calculation of these short-term and long-term dispersion values to be consistent with NRC guidance and, therefore, acceptable to be used in the calculation of radiological consequences with the understanding that the applicant will perform a short-term and long-term dispersion analysis with site-specific data to either: 1) affirm that dose calculations using the dispersion values in the PSAR are conservatively bounding, or 2) provide site-specific analysis results using dispersion factors developed during site characterization.

Therefore, based on its findings above, the NRC staff concludes that the information on meteorology in PSAR section 2.3 is sufficient and meets the applicable guidance and regulatory requirements for the issuance of a CP in accordance with 10 CFR 50.34(a)(1)(ii), 10 CFR 100.20(c)(2) and 10 CFR 100.21(d) and (e).

2.4 Hydrology

This section of the SE documents the NRC staff's review of the hydrological characteristics of the LMGS site and describes natural phenomena related to hydrology that have the potential to affect the LMGS design basis.

The applicable regulatory requirements related to the hydrological description of the site are as follows:

- 10 CFR 50.34(a)(1)(ii);
- 10 CFR 100.20; and
- 10 CFR 100.21.

The applicable guidance for the evaluation of the hydrological description of the site is:

- DANU-ISG-2022-02, Chapter 2, Section 2.5, "Hydrological Description." (ML23277A140);
- RG 1.59, "Design Basis Floods for Nuclear Power Plants." (ML003740388); and
- RG 1.102, "Flood Protection for Nuclear Power Plants." (ML003740308).

In accordance with 10 CFR 100.20(c) and 10 CFR 100.21(d), applications must contain a description of all site hydrological characteristics (e.g., probable maximum flood, ground water table, and aquifers) and summarize the design bases for the site characteristics selected for the design of SR and certain NSRST SSCs and the analysis of the transport of radioactive material resulting from postulated spills or leaks of liquid waste. Details of the hydrological information used to establish site characteristics or postulated site parameters may be documented in separate documents or reports that were made available for NRC staff audit.

The NRC staff's technical evaluation relied on the information presented in the PSAR. The NRC staff confirmed the information provided in the PSAR through audit discussions which are summarized in the audit summary report. The NRC staff used the information obtained during the regulatory audits to confirm the applicant's analytical methodologies, underlying

assumptions, data sources, and bases for the conclusions presented in the PSAR. The NRC staff did not perform independent confirmatory analyses.

This evaluation approach provides the foundation for the analyses presented in the following subsections, which address the technical bases, supporting data, and the NRC staff assessments associated with the applicant's hydrologic site characterization, flood hazard evaluations, and related site hydrology considerations.

2.4.1 Hydrologic Description

2.4.1.1 Technical Evaluation

The NRC staff's review of the information in this section of the PSAR focused on characterizing the governing hydrological processes at the LMGS site. The NRC staff confirmed that the information adequately describes the hydrological setting of the area, watershed, and basin. The NRC staff also reviewed the data and information to confirm the appropriateness of the data used in making safety conclusions.

The applicant described the hydrologic characteristics of the LMGS site, including the regional and local hydrologic setting, principal surface water features, water supply sources, drainage characteristics, and surrounding surface water and groundwater users. The information provides the hydrologic basis for the flood hazard evaluations presented in PSAR sections 2.4.2 through 2.4.13.

2.4.1.1.1 Site and Facilities

The applicant describes the LMGS site as a 1537-acre parcel located within Dow's Seadrift Operations property in Calhoun County, Texas. The NRC staff reviewed the applicant's characterization of site topography, surface-water features, and adjacent industrial and agricultural land uses. The site is traversed by West Coloma Creek, an existing stormwater channel conveying runoff from agricultural fields north of the property. The LMGS site location and topography are presented in PSAR figure 2.4.1-1 and figure 2.4.1-2, respectively.

The applicant confirmed that the NI site grade is 31.0 ft (9.4 m) North America Vertical Datum of 1988 (NAVD 88), and safety-related structures are proposed at a finished floor elevation of 31.5 ft (9.6 m) NAVD 88. Based on the site's elevation and distance of approximately 6 mi (10 km) from the Guadalupe River, the NRC staff finds the applicant's conclusion reasonable that channel diversions or riverine flooding would not pose a hazard to SR SSCs. The applicant identifies several man-made basins near the site and concludes they do not adversely affect flood potential, and the NRC staff agrees with this conclusion because the applicant identified all existing drainage pathways as it is supported by site mapping.

The NRC staff conducted multiple audit discussions to resolve outstanding questions, and the results of the discussion are summarized as follows:

Elevations of structures as confirmed during the audit discussions are provided in PSAR table 2.4.2.1-1. All elevations are in ft in reference to NAVD 88 and provided in PSAR (Revision 1) table 2.4.10-1. A summary of classification of buildings at the LMGS site is provided in PSAR table 2.4.2.1-2 (see also PSAR table 3.7-1).

2.4.1.1.2 Hydrosphere

The applicant provided an overview of major hydrologic features within the Lower Guadalupe River Basin, including the Guadalupe and San Antonio Rivers, Green Lake, West Coloma Creek, and multiple bays and canal systems. The NRC staff reviewed this information to obtain an adequate understanding of regional hydrology and its relationship with the site.

The applicant described the regional and local hydrologic setting of the LMGS site within the Lower Guadalupe River Basin. The applicant identified the principal hydrologic features surrounding the site, including the Guadalupe and San Antonio Rivers, Green Lake, Mission Lake, West Coloma Creek, the Guadalupe-Blanco River Authority (GBRA) Calhoun Canal System, the VBC, the San Antonio Bay system, wetlands, and other local drainage features.

The applicant described the watershed characteristics of the Guadalupe and San Antonio River Basins, including regional precipitation patterns, historical droughts and floods, streamflow records, upstream reservoirs and dams, and available United States Geological Survey (USGS) stream gaging stations used to evaluate hydrologic conditions and water availability. The applicant also described local wetlands, seasonal drainage features, the GBRA Calhoun Canal System, the raw water makeup (RWMU) system water supply, regional surface water users, water rights, and groundwater resources. The applicant stated that groundwater is not used to support plant operations and that the evaluations of water availability and potential impacts to surface water and groundwater users are presented in PSAR section 2.4.

The NRC staff reviewed the hydrologic descriptions contained in the PSAR, supporting hydrologic reports, maps, figures, and references, and clarifications provided by the applicant during the regulatory audits. The NRC staff finds that the applicant adequately characterized the regional and local hydrologic environment and identified the principal hydrologic features relevant to the site's flood hazard evaluations, water supply, and environmental interfaces. The NRC staff further finds that the applicant provided sufficient information to support the preliminary hydrologic characterization of the site for the CP review. Accordingly, the NRC staff determined the information presented in this subsection of the PSAR acceptable and sufficient for the purpose of the CP review.

<u>Guadalupe and San Antonio Rivers</u>

The applicant describes the contributing drainage areas of the Guadalupe (5953 mi^2) and San Antonio (4180 mi^2) River Basins, including major tributaries and historical hydrologic behavior. The NRC staff finds the applicant’s description of precipitation patterns, drought history, and extreme flood events adequate for assessing representative and bounding conditions. PSAR figure 2.4.2.1-3 presents the watersheds within the Guadalupe and San Antonio River Basin.

The NRC staff finds that the applicant appropriately discussed the largest recorded floods (1998 for Guadalupe River at Victoria; 1967 for the San Antonio River at Goliad) and the significance of the 2010–2015 and 1950s droughts. The NRC staff finds this historical context acceptable for identifying long-term flow variability.

The applicant described canal and levee systems, including the VBC and the Lower Guadalupe Saltwater Barrier and Diversion Dam (fabridam). The function of the fabridam in maintaining freshwater levels is adequately explained and supports downstream analyses of water

availability and salinity control. The fabridam is located near Tivoli, Texas, on the Guadalupe River. Originally built with inflatable rubber bladders, its primary function is to maintain headwater elevations to supply fresh water to Calhoun County and to prevent saltwater from migrating upstream during low-flow periods.

Streamflow gaging data from USGS stations at Victoria, Coleto Creek, McFaddin, and Tivoli are summarized, including periods of record and flow statistics. The NRC staff finds the use of upstream gauges for flow-availability analyses reasonable given known limitations of the Tivoli gauge during high-flow events.

Green Lake

Green Lake is described in the PSAR as a shallow (~4 ft (1.2 m)), natural freshwater lake fed by overbank flows from the Guadalupe River. The NRC staff reviewed these characteristics and finds them consistent with published hydrologic information. The NRC staff determined that the lake does not pose unique flooding or hydrodynamic hazards to the site.

West Coloma Creek

The applicant identifies West Coloma Creek as an intermittent stream draining agricultural lands before ultimately discharging to Matagorda Bay. The creek is located approximately 830 ft (253 m) east of the NI. The NRC staff finds that the applicant's mapping and classification of the creek provide adequate detail for evaluating local runoff and probable maximum precipitation (PMP)induced flooding.

Guadalupe–Lavaca Coastal Basin and San Antonio Bay System

The applicant describes the interconnected bay systems, tidal ranges, freshwater inflows, and sedimentation patterns. The NRC staff finds that the information accurately characterizes downstream hydrodynamic conditions and provides sufficient basis for analyzing coastal interactions.

Local Hydrologic Features

The NRC staff finds that minor drainage ditches, small seasonal streams, and nearby operating basins are documented in the PSAR. The NRC staff finds the description in the PSAR is complete for identifying local surface-water influences on site drainage and flooding.

Wetlands

The PSAR states that wetlands are limited to areas associated with West Coloma Creek and small adjacent basins. The NRC staff finds that the applicant's reliance on National Wetland Inventory mapping acceptable because it is consistent with the standard approach and provides adequate support for wetland characterization.

Water Supply and Canal System

The NRC staff reviewed the applicant's detailed description of the GBRA Calhoun Canal System, including the diversion structure, control dams, siphons beneath the VBC, pump

stations, intake lines, and industrial distribution pathways. The NRC staff finds the hydraulic description sufficient to establish the water-supply reliability and routing.

The applicant identifies in the PSAR the protection of the LMGS site water supply against saltwater intrusion and discusses operational responses during extreme surge events. The NRC staff finds this information adequate to support later evaluations in SE sections 2.4.11 and 2.4.13.

Surface Water Users

The applicant identifies in the PSAR the regional municipal, industrial, and irrigation water users with permitted surface-water withdrawals in the Lower Guadalupe and Lower San Antonio River Basins. The NRC staff reviewed the inventories from Texas Commission on Environmental Quality (TCEQ) and GBRA and finds them adequate to support impact analyses for both LMGS plant operations and accidental releases.

Groundwater and Users

The applicant states in the PSAR that groundwater is not used to support LMGS plant operations and references data from well inventory presented in later sections. The NRC staff finds this description sufficient for the purposes of PSAR section 2.4.1.

2.4.1.2 Conclusions

Based on its review, the NRC staff finds that the applicant adequately identified the principal hydrologic features affecting the site and described the site's interface with the surrounding hydrosphere. The NRC staff finds that the information is acceptable and sufficient for the purpose of the CP because it provides an adequate basis for the preliminary evaluation of site hydrology and supports the subsequent evaluations of flooding, water availability, and hydrologic hazards presented in PSAR section 2.4.

2.4.2 Floods

PSAR section 2.4.2 describes historical flooding by examining historical records of abnormally high- water levels in streams, rivers, waterways, lakes and other surface water features at the proposed LMGS site or in the region of the site. This section of the PSAR also summarizes and identifies individual types of flood-producing phenomena, and combinations of flood-producing phenomena, considered in establishing the flood design bases for SR and certain NSRST SSCs. The PSAR also covers the potential effects of local intense precipitation. Although topical information may appear in PSAR sections 2.4.3 through 2.4.7 and section 2.4.9, the types of events considered, and the controlling event are reviewed in this section.

2.4.2.1 Technical Evaluation

The NRC staff reviewed the information provided in PSAR section 2.4.2, Floods, in accordance with the guidance in NUREG-0800, Standard Review Plan (SRP) section 2.4.2, Floods, Regulatory Guide (RG) 1.206, and applicable NRC guidance for flood hazard evaluations. The NRC staff evaluated the applicant's identification of applicable flooding mechanisms, historical

flooding information, preliminary design-basis flood considerations, and the methodology used to evaluate local intense precipitation (LIP).

2.4.2.1.1 Flood History

PSAR section 2.4.2.1 states that the primary natural events that could cause flooding near the LMGS site include flooding from the Guadalupe and San Antonio Rivers, as well as Coleto Creek, hurricane-related storm surges from the Gulf of America (formerly known as Gulf of Mexico), and LIP affecting the site. The PSAR states that there are no known historical records in the surrounding area of flooding caused by ice-related events, tsunamis, channel diversions, or dam failures.

As part of its review, the NRC staff evaluated the applicant's identification of applicable flooding mechanisms, historical flooding information, preliminary design-basis flood considerations, and the methodology used to evaluate LIP.

The applicant summarized available USGS stream gauge records for the Guadalupe River, San Antonio River, and Coleto Creek and identified the major historical flood events affecting the region, including the 1967 flood on the San Antonio River and Coleto Creek and the October 1998 flood on the Guadalupe River. The applicant also described significant historical hurricane events affecting the Texas coast, including Hurricanes Carla and Harvey, and summarized observed storm surge elevations in the vicinity of the site.

The NRC staff determined that the applicant adequately characterized the historical flooding conditions relevant to the LMGS site and appropriately identified the principal historical flooding events and mechanisms considered in the development of the site's design-basis flood hazards. The NRC staff finds that the information is acceptable and sufficient for the purpose of the CP review because it provides an adequate historical basis supporting the flood hazard evaluations presented in PSAR section 2.4.

2.4.2.1.2 Flood Design Considerations

The applicant described the preliminary design-basis flood hazards considered for the LMGS site, including the probable maximum flood (PMF) on streams and rivers, dam failures, probable maximum hurricane surge and seiche, probable maximum tsunami, flooding due to ice effects, channel diversions, and LIP. The applicant stated in the PSAR that these flooding mechanisms were evaluated using combinations of coincident meteorological and hydrologic conditions consistent with the guidance in ANSI/ANS-2.8.

The applicant summarized in the PSAR the preliminary flood elevations associated with each flooding mechanism and identified probable maximum hurricane surge and seiche flooding as the controlling external flood hazard for the LMGS site. The applicant also identified preliminary flood levels associated with PMF, dam failures, LIP, and other flooding mechanisms, with detailed technical analyses provided in subsequent subsections of PSAR section 2.4.

The flood estimates as reported in Revision 1 of the PSAR include the following:

- LIP flood level is estimated to be 33.5 ft (10.1 m) NAVD 88 with a ponding depth of 2 ft (0.6 m) above the finished floor grade.

- The maximum PMF stillwater level over the Guadalupe River Basin combined with 500-year flow from the San Antonio River at the LMGS site, before wind-wave induced setup and runup, is predicted to be at elevation 28.0 ft (8.53 m) NAVD 88. Adding the combined maximum wind setup, wave runup predictions and sea-level rise, the maximum Gudalupe River PMF flooding water level at the LMGS site is postulated to be at elevation 40.80 ft (12.44 m) NAVD 88.

- The maximum PMF stillwater level of the San Antonio River Basin combined with 500-year flow from the Guadalupe River at the LMGS site, before wind-wave induced setup and runup, is predicted to be at elevation 26.0 ft (7.92 m) NAVD 88. Adding the combined maximum wind setup, wave runup predictions and sea-level rise, the maximum San Antonio River PMF flooding water level at the LMGS site is postulated to be at elevation 37.38 ft (11.39 m) NAVD 88.

- The maximum water surface elevation of flooding resulting from PMF on West Coloma Creek coincidental with wind setup and wave runup is estimated to be 33.56 ft (10.23 m) NAVD 88.

- The maximum water surface elevation of flooding from upstream dam failure is estimated to be 38.42 ft (11.71 m) NAVD 88 with stillwater and wave runup considerations.

- The maximum water surface elevation of flooding from failure of onsite water control or storage structures is estimated to be 32.5 ft (9.91 m) NAVD 88.

- Hurricane induced storm surge flooding is estimated to be 36.38 ft (11.1 m) NAVD 88 stillwater level, and including wave runup the maximum hurricane induced storm surge flooding is estimated to be 46.49 ft (14.17 m) NAVD 88.

- Tsunami flooding at the site is estimated to be 17.06 ft (5.2 m) NAVD 88.

These scenarios are analyzed by the applicant in accordance with ANSI/ANS-2.8-1992 guidelines, which recommend considering coincidental meteorological phenomena such as wind-generated waves and tidal variations.

A summary of the flood-causing mechanisms considered by the applicant and the resulting flood hazard level estimates evaluated by the NRC staff is presented in PSAR table 2.4.10-2.

2.4.2.1.3 Effects of Local Intense Precipitation

The applicant evaluated in the PSAR flooding resulting from LIP using PMP rainfall depths derived from Hydrometeorological Reports (HMR) 51 and 52 and the guidance provided in NUREG/CR-7046 “Design-Basis Flood Estimation for Site Characterization at Nuclear Power Plants in the United States of America” (ML11321A195). Because the final site grading and drainage design has not been completed, the applicant stated that the analysis was performed using conservative assumptions, including a flat NI grading configuration, fully blocked storm drainage systems, no infiltration, and unrestricted discharge from the site boundaries.

Probable Maximum Precipitation Depths

The applicant developed in the PSAR the LIP-PMP depths and temporal distribution for the LMGS site. The applicant developed in the PSAR the rainfall hyetographs for multiple storm durations to support the LIP evaluation. The PSAR stated the applicant's analysis performed to estimate LIP depth is based on estimating 6-hour PMP for the LMGS watershed containing the peak 1-hour storm. The LIP depth for the LMGS site is derived from U.S. NWS Hydrometeorological Reports No. 51 and 52, which provide all-season, 1 mi² point PMP depths and intensities. For durations up to one hour, point PMP depths reach 19.4 in. (493 mm) for 1 hour and 6.2 in. (157 mm) for 5 minutes. Intermediate durations such as 15 and 30 minutes were estimated by applying ratios from HMR 52 (NOAA, 1982), resulting in rainfall depths of 9.7 in. (24.6 cm) (15 min), 14.2 in. (36.1 cm) (60 min), and 32 in. (81.3 cm) (6 hr), respectively. These values are summarized in PSAR table 2.4.2-5. Using procedures from NUREG/CR-7046, a 5-minute interval PMP rainfall time series was constructed by the applicant with incremental precipitation depths based on these data.

The NRC staff reviewed the methodology, assumptions, and supporting documentation provided in the PSAR and during the regulatory audits. The NRC staff finds that the applicant's use of HMR 51 (NOAA, 1978), HMR 52, and NUREG/CR-7046 provides an acceptable preliminary basis for estimating PMP rainfall inputs for the CP review.

Local Drainage Components and Sub-Basins

The preliminary information in PSAR section 2.4.2.3.2 indicates that the LMGS NI will be constructed on fill material, raising the area approximately 4 ft (1.2 m) above existing grade. The preliminary layout of the LMGS NI is presented in PSAR figure 2.4.2-6.

The applicant described the preliminary hydraulic representation of the NI and stated that the final grading and drainage system have not yet been designed. Therefore, the applicant assumed a conservative site configuration that included a flat NI, inoperable storm drainage systems, and unrestricted runoff from the site boundaries.

The NRC staff noted that the applicant made the following assumptions for the preliminary LIP flood analysis:

1. The NI area is considered flat.

2. All storm drains, culverts, and catch basins are assumed to be clogged, rendering storm water collection and management systems inoperative.

3. No vehicle barriers or fencing are assumed, allowing natural drainage off the site on all sides.

The applicant stated that the LIP flood analysis is based on a two-dimensional hydrologic and hydraulic model which was developed to analyze drainage paths impacting flood levels near SR structures. Areas outside the NI were excluded from the analysis.

The NRC staff reviewed the assumptions used in the preliminary LIP analysis and the supporting hydraulic model documentation reviewed during the regulatory audits. The NRC staff finds the assumptions are appropriately conservative for a preliminary design evaluation and

provide an acceptable basis for the CP review. The final site configuration will be evaluated by the NRC staff during a review of the OL application.

Flood Elevations

The NRC staff reviewed the information provided in the PSAR including the results of the preliminary LIP flooding analysis. The applicant presented a description of the LIP flood model developed using the FLO-2D modeling software. Flood routing was modeled for the LMGS site using the FLO-2D two-dimensional model, which model's floodwater as flowing naturally without predefined channels. The model represents the site as a flat surface with buildings as obstructions and covers approximately 0.1 mi² (0.3 km^2) with a grid resolution of 5 ft (1.5 m) cells. The extent of the FLO-2D model and building locations is presented in PSAR figure 2.4.2-7.

The FLO-2D model developed by the applicant applied rainfall directly to each grid cell using the LIP distribution. Runoff was permitted to exit freely at model boundaries to replicate natural drainage. Manning's roughness coefficient of 0.04 was assigned in the model to represent crushed compact stone ground cover, and soil infiltration was neglected, consistent with NUREG/CR-7046 guidelines. Buildings and other structures were incorporated into the model using area reduction factors and width reduction factors derived from shape files. For the buildings, a Manning's roughness coefficient of 0.01 was used in the model.

To simulate proper drainage from roofs in the model, elevations of grid cells representing building roofs and tanks were raised by 3 ft (0.9 m). Vehicle barrier systems were not included in the preliminary site layout and, therefore, not considered in the model. The NRC staff determined that the FLO-2D model results indicate a maximum flood depth of approximately 24 inches (0.6 m) above finished grade at building locations near SR doors. Flood waters are predicted to remain above the top of foundation elevation for approximately 12 hours, as shown in the representative PMP flood depth time series PSAR figure 2.4.2-8. As reported in the PSAR, maximum floodwater velocities during the event remain below 0.5 ft/s (0.15 m/s), indicating minimal erosion potential at the site.

The NRC staff determined that the technical approach followed in the model to characterize flood-causing mechanism is appropriate because it follows the applicable guidelines provided in NUREG/CR-7046 and the FLO-2D User's Manual for characterization and modeling of the flood-causing mechanism. The NRC staff reviewed the applicant's 5-ft (1.5 m) model cell refinement and determined that the selected 5-ft (1.5 m) cell size provides sufficient spatial resolution for the FLO-2D simulation and adequately represents the LIP flow characteristics relevant to the site evaluation. The NRC staff also determined the assumptions used by the applicant to estimate LIP flooding at the LMGS site are conservative because the analysis uses a conservative precipitation scenario and assumes blockage of onsite drainage. The LIP flooding analysis results presented in the PSAR indicate that the LIP flood depth exceeds finished grade level by 24 in (0.6 m). The applicant reported that the maximum LIP-induced flood velocity near the entrance door to the SR buildings is less than 0.5 ft/s (0.15 m/s). The NRC staff reviewed the LIP flooding analysis report in audit to confirm the information provided in the PSAR and identified areas within the FLO-2D model domain, away from the entrance door, where the flow velocity approaches 2 ft/s (0.6 m/s) and other areas where it exceeds 2 ft/s (0.6 m/s). The velocities induced by LIP will also be impacted by the site grading and any passive engineered flood protection barriers constructed at the site. However, the site grading and passive engineered flood protection barrier design were not provided at the CP stage as

required as part of meeting 10 CFR 100.21(d). For these two aspects, consistent with 10 CFR 50.35(a), the NRC staff finds that the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public and these safety aspects will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility. The NRC staff proposes the following site grading and passive engineered flooding protection design permit condition:

> LME shall notify the Director of the Office of Advanced Reactors, or the Director's designee, as specified in 10 CFR 50.4, when the site grading design and the passive engineered flooding protection design that may impact site grading are available for examination by the NRC staff. This is expected to occur prior to the placement of safety-significant concrete for structure foundations. The design should consider the primary site flood hazard causing mechanisms described in PSAR section 2.4. The design should also demonstrate that safety-related structures, systems, and components (SSCs) and the subset of non-safety-related with special treatment SSCs that must remain functional during and after the design basis flood event are protected. Aspects of site grading and engineered flooding protection features that may impact the results of previous flooding analyses (i.e., design inputs and assumptions) should be addressed such that the design and analyses are consistent.

Per 10 CFR 50.34(b)(1), each application for an OL shall include a final safety analysis report (FSAR), and the FSAR shall include "All current information, such as the results of environmental and meteorological monitoring programs, which has been developed since issuance of the CP, relating to site evaluation factors identified in 10 CFR Part 100. Further, per 10 CFR 50.34(b)(11), stationary power reactor applicants who apply for an OL, shall provide a description and safety assessment of the site and of the facility as in 10 CFR 50.34(a)(1)(ii), which includes in 10 CFR 50.34(a)(1)(ii)(D) that site characteristics must comply with Part 100 of this chapter.

2.4.2.2 Conclusions

Based on its review of the PSAR, and supporting information reviewed during the regulatory audits, the NRC staff finds that the applicant identified the flooding mechanisms applicable to the LMGS site and provided an adequate technical basis for the preliminary flood hazard evaluations presented in PSAR section 2.4. The NRC staff finds the information acceptable and sufficient for the purpose of the CP review because it establishes the basis for the subsequent detailed flood hazard evaluations presented in PSAR sections 2.4.3 through 2.4.10.

The NRC staff concludes that the analyses provide reasonable assurance that the bounding flood hazard and associated flood effects do not present a siting issue that would preclude construction of the proposed LMGS facility.

The NRC staff notes that the LIP flood analyses are based, in part, on assumptions for site grading and drainage configurations because the site grading design, flood protection features, and detailed arrangement of the safety-significant structures and shared SSCs have not been finalized. The site grading and passive engineered flooding protection design permit condition requires that LME notify the NRC staff once these aspects of the design and analysis are available.

2.4.3 Probable Maximum Flood in Streams and Rivers

PSAR section 2.4.3 describes flooding at the LMGS site resulting from PMP over the watershed and basin in which the proposed plant is located. The NRC staff evaluated whether the application provides the necessary information and technical analysis describing the watershed and basins considered and the PMP used to create the basin/watershed flood model. The application should describe the hydrometeorological design basis used to determine the extent of any flood protection required for SR and certain NSRST SSC necessary to ensure safety.

2.4.3.1 Technical Evaluation

The applicant described the hydrologic setting of the LMGS site and evaluated the potential for flooding from the Guadalupe River and West Coloma Creek. Although the LMGS site is located outside the Guadalupe River floodplain, the applicant identified the potential for floodwaters to overtop the eastern bluff and affect the LMGS site. The LMGS site grade ranges from approximately 27 to 29 ft (8.2 to 8.8 m) NAVD 88, while the finished floor elevation of all SR structures is established at 31.5 ft (9.6 m) NAVD 88. PSAR table 2.4.10-1 provides the finished floor elevation for all SR and NSRST structures as 31.5 ft (9.6 m), apart from the spent fuel intermediate storage facility which has a finished floor elevation of 27.4 ft (8.4 m) NAVD 88.

The applicant identified the Guadalupe River watershed upstream of the site as encompassing approximately 5,953 mi² (15,418 km²) and documented the presence of 29 upstream dams with storage capacities greater than 3,000 acre-feet (3.7 million m^3). The applicant identified Canyon Dam and Coleto Creek Dam as the principal flood-control structures influencing hydrologic conditions due to their significant storage capacities. Information regarding these dams is provided in PSAR sections 2.4.1 and 2.4.4.

The applicant separately evaluated flooding from West Coloma Creek because the LMGS site is located within its watershed. Flooding associated with LIP and dam failures is addressed separately in sections 2.4.2 and 2.4.4, respectively.

Consistent with ANSI/ANS-2.8-1992, the applicant assumed in its analysis that PMP over the Guadalupe River watershed would occur concurrently with an appropriate river discharge rather than simultaneously with PMP over both the Guadalupe and San Antonio River Basins. The applicant also assumed that snowmelt and antecedent snowpack are not credible contributors to PMF at the LMGS site because of the very low probability of snowfall within the Lower Guadalupe River Basin. Additionally, the applicant assumed that urbanization within the watershed would not significantly affect flood runoff characteristics because of the large drainage area.

The applicant based the hydrologic evaluation on numerous federal, state, and local studies, including investigations performed by the U.S. Army Corps of Engineers (USACE), GBRA, FEMA, Texas Water Development Board (TWDB, 1980 and 2021), San Antonio River Authority (SARA), and the Interagency Flood Risk Management (InFRM) program. These studies provided the basis for development and refinement of hydrologic and hydraulic models representing the Guadalupe and San Antonio River watersheds.

The PSAR states that the Guadalupe River Basin, draining 5,953 mi² (15,418 km^2), is heavily regulated by 29 major flood-control structures. The PSAR identifies Canyon Dam (top elevation 974.34 ft (297 m) NAVD 88; flood storage 1.21 million acre-ft (1.5 billion m^3)) and Coleto Creek

Dam (top elevation 119.71 ft (36.49 m) NAVD 88; flood storage 149,800 acre-ft (185 million m^3)) as the most significant hydrologic structures, which significantly reduce flood peaks reaching the LMGS area.

The NRC staff finds that the applicant has appropriately identified the hydrologic features that could influence stream and river flooding at the LMGS site, including the Guadalupe River, West Coloma Creek, and significant upstream reservoirs. The applicant's use of updated watershed models, reliance on established federal and state hydrologic studies, and application of ANSI/ANS-2.8-1992 guidance provide an acceptable basis for the subsequent PMF analyses presented in PSAR section 2.4.3. The NRC staff finds that exclusion of snowpack-related PMF alternatives is technically justified based on the climatic conditions in the vicinity of the LMGS site. The NRC staff concludes that the information provided establishes an acceptable basis for evaluating floods on streams and rivers for the CP application, subject to the detailed hydrologic and hydraulic results presented in the following subsections.

2.4.3.1.1 Design Bases for Flooding in Streams and Rivers

The PSAR evaluates the effects of flooding at the LMGS site resulting from the PMF in the Lower Guadalupe River and West Coloma Creek. Flooding from LIP and potential dam failures is addressed separately in PSAR sections 2.4.2 and 2.4.4, respectively. Because no prior flood study was identified for West Coloma Creek, the applicant performed an independent analysis, as described in PSAR subsection 2.4.3.3.

The applicant determined that snowmelt and antecedent snowpack are not significant contributors to flooding at the LMGS site based on the air temperature characteristics of the Lower Guadalupe River Basin. The applicant also concluded that urban development is not expected to materially affect flood runoff or flood levels at the site because of the large Guadalupe River drainage area of approximately 5,953 mi^2 (15,418 km^2). Conservative assumptions were used in evaluating the West Coloma Creek watershed to address limitations in available local data.

The applicant reviewed hydrologic and hydraulic studies developed by the U.S. Army Corps of Engineers (USACE), FEMA, SARA, TWDB, GBRA, and InFRM, including studies addressing Canyon Lake, Coleto Creek Dam, the Guadalupe River Basin, and regional watershed hydrology. Of particular relevance were: (1) USACE studies of Canyon Lake, representing a drainage area of approximately 1,425 mi^2 (3,691 km^2); (2) Halff Associates' studies of Coleto Creek Dam, representing approximately 491 mi^2 (1,272 km^2); and (3) the 2019 InFRM Guadalupe River Basin watershed assessment, which incorporated updated hydrologic modeling and calibration.

The applicant used these existing studies as a basis for development of the design-controlling PMF for the LMGS site. The existing analyses were independently adapted by the applicant to incorporate more recent observations and modeling information for the Guadalupe and San Antonio River watersheds. The resulting modeling was used to evaluate the flood hazard associated with precipitation events in accordance with ANSI/ANS-2.8-1992, including the specified alternative combinations of baseflow, soil moisture or snowpack, antecedent precipitation, and probable maximum precipitation/snow-season precipitation.

Because the LMGS site is located in an area where snow-related flood mechanisms are not considered significant, the applicant excluded the ANSI/ANS-2.8-1992 alternatives requiring probable maximum snowpack or 100-year snowpack. Consequently, the snow-season alternatives were not considered applicable to the LMGS site.

The NRC staff determined that the PSAR establishes the design basis for river and stream flooding by evaluating the PMF in the Lower Guadalupe River and West Coloma Creek, using applicable regional hydrologic studies supplemented by independent analyses and conservative assumptions. The NRC staff agrees with the applicant's determination that snowmelt and snowpack are not significant flood-producing mechanisms provides the basis for limiting the ANSI/ANS-2.8-1992 evaluation to the applicable precipitation-driven flooding scenario.

2.4.3.1.2 Identification and Analysis of the Probable Maximum Flood

The applicant established in the PSAR the design basis for stream and river flooding by developing independent hydrologic analyses of the Guadalupe River and San Antonio River Basins using updated Hydrologic Engineering Center–Hydrologic Modeling System (HEC-HMS) models. The applicant also stated that the analyses incorporated the latest available hydrologic observations and modeling improvements to estimate the PMF affecting the LMGS site.

Consistent with ANSI/ANS-2.8-1992, the applicant assumed that simultaneous PMP over both the Guadalupe and San Antonio River watersheds is not a credible event. Instead, the applicant modeled the controlling watershed under PMP conditions concurrent with the appropriate downstream river discharge and a 500-year rainfall event over the non-controlling watershed.

The NRC staff reviewed the applicant’s hydrologic approach, including the treatment of PMP in the two watersheds and the use of the controlling watershed with a 500-year rainfall event in the non-controlling watershed. The NRC staff determined that the approach provides a reasonable and technically appropriate basis for estimating the PMF at the LMGS site and is acceptable for the CP review.

Guadalupe River Basin

The applicant developed the Guadalupe River Basin runoff model using the HEC-HMS platform based primarily on the U.S. Army Corps of Engineers (USACE) Corps Water Management System (CWMS) model and subsequent refinements performed under the InFRM program. The applicant stated that the InFRM model expanded the watershed representation from 72 to 165 subbasins, incorporated the Coleto Creek watershed, refined watershed delineations, updated routing methods, and improved representation of hydraulic control structures and reservoirs.

The applicant described the geospatial datasets used to develop the watershed model, including LiDAR-derived digital elevation models referenced to NAD 1983 and NAVD 88. The applicant further stated that watershed parameters, including precipitation losses, runoff transformation methods, baseflow representation, stream routing, and reservoir storage-routing relationships, were updated using current hydrologic information.

The applicant identified Canyon Lake Reservoir as the principal flood-control reservoir influencing downstream flood response. Reservoir behavior was represented using storage-

discharge and elevation-storage relationships developed during the InFRM assessment to simulate reservoir storage and release during extreme hydrologic events.

To establish model validity, the applicant independently recalibrated the HEC-HMS model using observed precipitation and streamflow data from the October 1998 flood event. Calibration parameters included watershed losses, time of concentration, and routing coefficients. Model performance was evaluated using the Nash-Sutcliffe Efficiency (NSE) statistics. The applicant reported NSE values ranging from 0.71 to 0.99 at the calibration gauges, with simulated peak discharges conservatively exceeding observed peak flows. The applicant concluded that the calibrated model provides an appropriate and conservative basis for extrapolation to PMF conditions.

San Antonio River Basin

The applicant developed the San Antonio River Basin HEC-HMS model by integrating previously calibrated hydrologic models developed by the SARA, TWDB, and other regional studies with additional watershed expansion and calibration performed specifically for this application.

The applicant modeled the watershed from its headwaters to the confluence with the Guadalupe River, representing 61 subbasins, 43 stream reaches, and six major reservoirs. Watershed delineation was developed using USGS digital elevation models, while land use and soil characteristics were incorporated through National Land Cover Dataset and USDA Natural Resources Conservation Service (NRCS) datasets.

The applicant used the Soil Conservation Service (SCS) curve number method to estimate infiltration losses and the Modified Clark (ModClark) unit hydrograph method to transform rainfall excess into runoff. Stream routing employed Muskingum, Muskingum-Cunge, and Modified Puls methods, consistent with previously developed SARA watershed models. Reservoir routing incorporated storage-discharge and elevation-storage relationships for the principal flood-control reservoirs identified within the basin.

The applicant calibrated the San Antonio River model using the October 1998 flood event and observed streamflow records from four USGS gaging stations distributed throughout the watershed. Calibration focused on optimizing watershed loss parameters and routing coefficients using the NSE statistics. The applicant reported NSE values ranging from 0.80 to 0.83, indicating acceptable agreement between simulated and observed hydrographs for application to PMF analyses.

The NRC staff finds that the applicant developed hydrologic models for both the Guadalupe River and San Antonio River Basins using accepted HEC-HMS methodologies supported by current topographic, hydrologic, land use, and soil datasets. The NRC staff determined that the applicant appropriately incorporated previously validated regional watershed models and performed independent recalibration using the October 1998 extreme flood event to confirm model performance under severe hydrologic conditions. The NRC staff finds the reported calibration statistics and conservative reproduction of observed peak discharges provide reasonable assurance that the models are suitable for estimating PMF inflows used in subsequent flood hazard analyses. The NRC staff concludes that the hydrologic modeling methodology and model calibration provide an acceptable technical basis for establishing the

design-basis floods on streams and rivers for the CP application, subject to review of the resulting PMF calculations and hydraulic analyses presented in the following sections.

Guadalupe River Basin Discharge

The applicant developed a hydrologic model of the Guadalupe River Basin to estimate runoff for the PMF analysis using the HEC-HMS. The applicant stated that the model represents watershed characteristics through subbasin delineation, precipitation loss methods, rainfall-runoff transformation, stream routing, and reservoir routing.

The applicant based the initial model on the U.S. Army Corps of Engineers (USACE) CWMS HEC-HMS model and subsequently incorporated refinements developed through the 2019 InFRM Guadalupe River Basin Watershed Hydrology Assessment. The applicant stated that these previously validated models were further updated to incorporate current hydrologic observations and improve representation of extreme flood events.

The applicant described the geospatial datasets used to construct the watershed model, including LiDAR-derived digital elevation models referenced to NAD 1983 and NAVD 88. The applicant stated that 10-meter digital elevation models were used for watershed delineation and 1-meter digital elevation models were used for hydraulic analyses.

The applicant explained that the original CWMS model represented approximately 5,198 m^2 (13,463 km^2) using 72 subbasins but excluded the Coleto Creek watershed and portions of the Lower Guadalupe River Basin. The applicant adopted the expanded InFRM model, which increased the watershed representation to 165 subbasins, incorporated the Coleto Creek watershed, and refined subbasin boundaries throughout the basin to improve representation of tributaries, reservoirs, hydropower facilities, and flood routing.
The applicant stated that the watershed model utilizes accepted HEC-HMS methodologies, including deficit and constant precipitation loss, Snyder Unit Hydrographs for rainfall-runoff transformation, recession baseflow, and Modified Puls routing for reservoirs. The applicant further stated that routing parameters and hydraulic relationships were updated using LiDAR-based HEC-RAS models, while Muskingum routing was retained for portions of the Coleto Creek watershed where detailed hydraulic information was unavailable. The applicant also reduced the computational time step from 60 minutes to 15 minutes to improve temporal resolution during extreme storm simulations.

The applicant obtained historical precipitation inputs from NWS NEXRAD Stage III precipitation datasets and processed the data using HEC-MetVue to develop calibration rainfall inputs for the October 1998 flood event.

The applicant identified Canyon Lake Reservoir as the principal flood-control feature within the upper Guadalupe River Basin. The applicant represented reservoir operation using empirical storage-discharge and elevation-storage relationships developed during the InFRM study. Although the physical outlet works of Canyon Dam were not explicitly modeled, the applicant stated that these functional relationships adequately reproduce reservoir storage and discharge behavior under extreme hydrologic conditions.

To validate the model, the applicant independently recalibrated the InFRM HEC-HMS framework using observed rainfall and streamflow records from the October 1998 flood event. Calibration included adjustment of watershed losses, time of concentration, and routing

parameters using the HEC-HMS optimization tool. The applicant evaluated model performance using the NSE statistics and reported values ranging from 0.71 to 0.99 at the calibration gauges. The applicant further stated that simulated peak discharges consistently exceeded observed peak flows, thereby providing a conservative basis for PMF estimation.

The NRC staff finds that the applicant developed the Guadalupe River Basin hydrologic model using accepted HEC-HMS methodologies and based the model on previously validated USACE and InFRM watershed studies. The NRC staff finds that the applicant appropriately updated watershed geometry, routing methods, reservoir representation, and precipitation inputs using current datasets and independently recalibrated the model using the October 1998 historical flood event. The NRC staff determined the reported calibration statistics demonstrate acceptable agreement with observed hydrologic conditions, while the conservative overprediction of peak discharges provides reasonable assurance that the model is suitable for estimating design-basis PMF runoff. Accordingly, the NRC staff concludes that the Guadalupe River Basin hydrologic model provides an acceptable technical basis for subsequent hydraulic analyses supporting the CP application.

San Antonio River Basin Discharge

The applicant developed a hydrologic model of the San Antonio River Basin using HEC-HMS to estimate runoff contributions from the basin during design-basis flood events. The applicant stated that the model incorporates watershed characteristics through subbasin delineation, precipitation loss calculations, rainfall-runoff transformation, stream routing, and reservoir routing.

The applicant's model was developed by integrating previously calibrated hydrologic models prepared by the SARA, TWDB, and other regional studies with additional watershed expansion and independent calibration performed specifically for the CP application.

The applicant stated that the watershed model extends from the headwaters of the San Antonio River Basin to its confluence with the Guadalupe River and represents 61 subbasins, 43 stream reaches, and six major reservoirs. The applicant utilized USGS digital elevation models to define watershed boundaries and incorporated existing FEMA Flood Insurance Study information and SARA hydrologic datasets where appropriate.

The applicant incorporated watershed soil and land use characteristics using USDA NRCS Hydrologic Soil Groups and NLCD information. The applicant developed spatially varying SCS curve numbers from these datasets to estimate infiltration losses throughout the watershed. The applicant noted that clay-rich soils within the lower basin and urban development within the upper basin contribute to rapid runoff and elevated peak discharges during major precipitation events.

The applicant selected the Modified Clark (ModClark) unit hydrograph method for rainfall-runoff transformation and employed Muskingum, Muskingum-Cunge, and Modified Puls routing methods to simulate downstream flood propagation. Routing parameters were obtained from previously calibrated SARA watershed studies and further refined during model calibration.

The applicant modeled the principal reservoirs within the watershed using storage-discharge, elevation-storage, and elevation-area relationships derived from existing calibrated models and National Inventory of Dams information. The applicant identified Medina Lake as the reservoir

having the greatest influence on downstream flood attenuation, while noting that Olmos and Calaveras Dams provide localized moderation of flood peaks generated within the upper urbanized watershed.

The applicant calibrated the watershed model using observed precipitation from the October 1998 flood event and streamflow data collected at four USGS stream gaging stations distributed throughout the watershed. Calibration focused on adjustment of watershed losses and routing parameters to maximize the NSE statistic. The applicant reported final NSE values ranging from 0.80 to 0.83, indicating acceptable agreement between simulated and observed hydrographs.

The NRC staff finds that the applicant developed the San Antonio River Basin hydrologic model using accepted HEC-HMS methodologies supported by current topographic, soil, land use, reservoir, and streamflow datasets. The NRC staff determined that the applicant appropriately incorporated previously validated regional watershed models and performed independent calibration using a historically significant flood event representative of severe basin response. The NRC staff determined that the reported calibration statistics demonstrate acceptable model performance for estimating runoff under extreme precipitation conditions. Accordingly, the NRC staff concludes that the San Antonio River Basin hydrologic model provides an acceptable technical basis for estimating tributary inflows to the Guadalupe River and for supporting the subsequent PMF and hydraulic analyses presented in PSAR section 2.4.3.

Probable Maximum Precipitation

The applicant established the PMP inputs for the PMF analyses using the methodologies contained in NWS Hydrometeorological Reports (HMRs) No. 51 and No. 52. The applicant stated that basin-average PMP values derived from the HMRs were compared with those developed by the TCEQ. Because the HMR-derived precipitation depths exceeded those developed in the TCEQ study, the applicant concluded that use of the HMR values provides a conservative basis for PMF analyses.

The applicant stated that HMR Nos. 51 and 52 were implemented using the HEC-HMS PMP meteorological model, which generates subbasin-specific precipitation hyetographs based on location-dependent depth-area-duration (DAD) relationships. To identify the most limiting meteorological configuration, the applicant evaluated multiple storm-center locations and storm orientations for both the Guadalupe River and San Antonio River Basins. The applicant selected four storm-center configurations for the Guadalupe River Basin and three configurations for the San Antonio River Basin to identify the storm placement producing the maximum hydrologic response at the LMGS site.

The applicant developed DAD relationships for each storm configuration using HMR No. 51 and expanded the analysis by interpolating precipitation depths for additional basin sizes surrounding the watershed drainage areas. The applicant stated that this sensitivity analysis is consistent with HMR guidance for identifying the controlling storm area that produces the maximum basin-average precipitation.

The applicant incorporated additional meteorological parameters required by HMR No. 52, including basin-specific one-hour to six-hour precipitation ratios. Consistent with ANSI/ANS-2.8-1992, the applicant also modeled antecedent precipitation equal to 40 percent of the 72-hour

PMP occurring three days before the design PMP event to establish conservative antecedent watershed moisture conditions.

The applicant utilized the HEC-HMS optimization tool to evaluate multiple combinations of storm size, location, and orientation. The optimization process identified the storm areas producing the maximum discharge at the model outfalls for both the Guadalupe River and San Antonio River Basins.

The NRC staff finds that the applicant developed the PMP inputs using accepted methodologies contained in NWS HMR Nos. 51 and 52 and implemented these methods through the HEC-HMS PMP modeling framework. The NRC staff determined that the applicant appropriately evaluated multiple storm locations, orientations, and basin sizes to identify the hydrologically controlling storm configuration and incorporated conservative antecedent moisture conditions consistent with ANSI/ANS-2.8-1992. The NRC staff concludes that the applicant's PMP development provides an acceptable and conservative meteorological basis for subsequent PMF analyses supporting the CP application.

Guadalupe River Hydraulics

The applicant evaluated flood elevations along the Guadalupe River using a one-dimensional HEC-RAS hydraulic model based on the U.S. Army Corps of Engineers (USACE) Lower Guadalupe Feasibility Study. The applicant stated that the hydraulic model represents approximately 296 miles (476 km) of the Guadalupe River extending from Canyon Dam to the Victoria-Calhoun County boundary near Tivoli, Texas, and incorporates hydraulic models for the Guadalupe, Blanco, and San Marcos Rivers.

The applicant stated that the hydraulic model incorporates LiDAR-derived topographic data, surveyed channel cross sections, bridge and culvert geometry, field measurements, as-built drawings, and existing FEMA hydraulic models. The applicant further stated that channel roughness coefficients, ineffective flow areas, blocked obstructions, and hydraulic structures were developed consistently with accepted HEC-RAS engineering practices and the HEC-RAS Hydraulic Reference Manual.

The applicant excluded the Lower Guadalupe Salt Water Barrier and Diversion Dam (Fabridam) from the PMF hydraulic model because the inflatable structure is designed to fail under hydraulic heads substantially below those expected during the PMF. The applicant concluded that failure of the structure would occur prior to the PMF peak and therefore would not influence maximum flood elevations.

The applicant represented major tributary confluences and complex floodplain interactions using one-dimensional storage areas connected through lateral structures. The applicant stated that this modeling approach was developed during the USACE Lower Guadalupe Feasibility Study to simulate backwater effects and multidirectional floodplain storage while maintaining computational efficiency. Storage-area connections were based on surveyed hydraulic features, including roadway crossings and natural terrain controls.

The applicant independently recalibrated the USACE HEC-RAS model using observed high-water marks from the October 1998 flood event. Calibration utilized hydrographs generated by the independently calibrated HEC-HMS watershed model together with 27 hydraulic boundary conditions representing inflows from Canyon Dam and major tributaries. The applicant modeled

Canyon Dam releases using minimum operational discharges consistent with historical reservoir operations during the October 1998 flood event.

The applicant validated the hydraulic model using observed USGS stream gauge elevations, FEMA Flood Insurance Study information, historical flood records, and the InFRM Flood Decision Support Toolbox. The applicant stated that the recalibrated model successfully reproduced the recorded peak stage of the October 1998 flood at the Cuero stream gauge and demonstrated improved agreement with observed flood elevations compared to the original USACE model. The applicant reported that the recalibration primarily focused on reproducing maximum flood depths along the Guadalupe River profile and at selected control gaging stations.

The NRC staff finds that the applicant developed the Guadalupe River hydraulic model using an established USACE HEC-RAS framework supported by current topographic data, surveyed hydraulic geometry, and accepted hydraulic engineering practices. The NRC staff determined that the applicant appropriately represented hydraulic structures, floodplain storage, tributary inflows, and reservoir operations and independently recalibrated the model using observed data from the October 1998 flood event. The NRC staff finds that the applicant's use of multiple historical datasets and observed high-water marks provides reasonable assurance that the hydraulic model adequately represents flood propagation and water surface elevations along the Guadalupe River during extreme flood events. Accordingly, the NRC staff concludes that the Guadalupe River hydraulic model provides an acceptable technical basis for determining design-basis flood elevations at the LMGS site for the CP application.

Water Level at the New Plant Location

The applicant evaluated the design-basis water level at the LMGS site by combining hydrologic results from the Guadalupe River and San Antonio River HEC-HMS models with the previously developed Guadalupe River HEC-RAS hydraulic model. The applicant stated that because the San Antonio River joins the Guadalupe River upstream of the site, the Guadalupe River hydraulic model was used to simulate the combined river response at the plant location.

Consistent with ANSI/ANS-2.8-1992, the applicant did not assume simultaneous PMP over both watersheds. Instead, the applicant evaluated the controlling watershed under PMP conditions while representing the adjacent watershed using either 100-year or 500-year rainfall events, depending on the stage of the simulation. During the antecedent 40 percent PMP phase, the neighboring watershed was represented by the 100-year event, while the full PMP simulation used concurrent 500-year flows from the adjacent watershed.

The applicant generated 72-hour hydrographs for the PMP, 100-year, and 500-year events using the calibrated HEC-HMS watershed models. The applicant synchronized the hydrographs such that peak flows from the controlling watershed coincided with the frequency-based peak flows from the adjacent watershed. These hydrographs were used as inflow boundary conditions for the unsteady HEC-RAS hydraulic simulations.

The applicant performed each PMF simulation in two stages. The first simulation established antecedent hydraulic conditions using the 40 percent PMP event, while the second simulation modeled the full PMP event beginning from the antecedent conditions through the HEC-RAS restart capability. The applicant stated that this approach provided continuous simulation of

watershed response over the 16-day analysis period and preserved the temporal relationship between antecedent rainfall and the controlling storm event.

The applicant evaluated the resulting water surface elevations at HEC-RAS River Station 1.255, corresponding to the vicinity of the LMGS site. The applicant identified the maximum computed water surface elevation from the simulated scenarios as the design-basis PMF water level for subsequent flood hazard evaluations.

The NRC staff finds that the applicant used an acceptable methodology to combine the independently developed Guadalupe River and San Antonio River hydrologic models within the calibrated Guadalupe River hydraulic model to determine design-basis water levels at the LMGS site. The NRC staff determined that the applicant appropriately implemented ANSI/ANS-2.8-1992 guidance by evaluating PMP on the controlling watershed concurrent with frequency-based flooding on the adjacent watershed and by accounting for antecedent moisture conditions. The NRC staff concludes that the hydraulic simulations provide an acceptable basis for determining PMF water surface elevations at the proposed plant location for the CP application.

Coincident Wave Runup

The applicant evaluated the effects of wind-generated waves coincident with the PMF to determine whether wave runup, wind setup, overtopping, or erosion could challenge flood protection at the LMGS site. The applicant performed the analysis using the Automated Coastal Engineering System (ACES) within the Coastal Engineering Design and Analysis System (CEDAS), consistent with U.S. Army Corps of Engineers (USACE) coastal engineering guidance and ANSI/ANS-2.8-1992.

The applicant stated that West Coloma Creek is normally dry, has no continuous baseflow, and functions primarily as an agricultural drainage channel. Accordingly, the applicant did not include baseflow in the wave generation analysis.

The applicant developed wave characteristics using site-specific hydraulic conditions calculated from the HEC-RAS PMF analysis. Inputs included restricted-fetch geometry, water depth, wind speed, wind duration, and fetch length. The applicant evaluated fifty potential fetch alignments to identify the controlling wind direction and fetch geometry. Wind speeds were developed from the ANSI/ANS-2.8-1992 design wind of 50 mph (80 kph) and converted to appropriate averaging durations using USACE Coastal Engineering Manual procedures.

The applicant selected the shallow restricted-fetch option within ACES because the floodplain geometry does not support fully developed deep-water waves. The applicant calculated significant wave height, wave period, design wave height (H1%), wind setup, and wave runup for each fetch. Wave runup was determined using USACE Coastal Engineering Manual guidance with the ANSI/ANS-2.8-recommended H1% wave height substituted for the significant wave height. The applicant combined the calculated wave runup and wind setup with PMF stillwater elevations to determine total water levels.

The applicant reported that computed wave runup ranged from approximately 0.5 to 5.85 ft (0.15 to 1.78 m) and wind setup ranged from approximately 0.19 to 0.87 ft (0.06 to 0.27 m). The applicant stated that the maximum wave runup occurs along the left bank of the Guadalupe River more than one mile (1.6 km) west of the LMGS site rather than at the site boundary.

The applicant evaluated intervening topographic and man-made features between the river and the LMGS site, including transportation embankments, industrial facilities, roadway crossings, natural depressional storage areas, and Highway 185. The applicant concluded that these elevated features function as hydraulic controls that intercept, store, or redirect any localized overtopping away from the plant site. The applicant further developed total water level inundation maps demonstrating that wave runup effects do not extend to the LMGS site.

For the limited riverbank reaches where overtopping was predicted, the applicant calculated overtopping discharge volumes using USACE Coastal Engineering Manual methods. The applicant conservatively assumed the maximum design wind speed acting over the longest calculated wind duration and estimated a maximum overtopping volume of approximately 2.3 million gallons (8.7 million liters). The applicant incorporated this overtopping volume into local hydraulic analyses and concluded that downstream roadways and intervening embankments would retain the overtopped water without permitting floodwater to propagate toward the LMGS site.

The applicant also evaluated potential embankment erosion caused by wave action using USACE Engineering Research and Development Center guidance. The applicant calculated maximum, average, and minimum erosion depths of approximately 0.56 ft (0.17 m), 0.12 ft (0.037 m), and 0.01 ft (0.003 m), respectively, and concluded that the predicted erosion would not compromise embankment integrity or produce a breach during the design-basis event.

The NRC staff finds that the applicant evaluated coincident wave effects using accepted coastal engineering methodologies consistent with ANSI/ANS-2.8-1992 and applicable USACE guidance. The NRC staff determined that the applicant appropriately assessed wave generation, wind setup, wave runup, overtopping potential, and embankment erosion using PMF hydraulic conditions and conservative design wind assumptions. The applicant further demonstrated that intervening terrain, engineered infrastructure, and natural drainage characteristics prevent wave runup or overtopping from reaching the LMGS site. Based on the analyses presented, the NRC staff concludes that coincident wave effects do not challenge the flood protection capability of SR structures at the proposed site and that the evaluation provides an acceptable basis for the CP application.

Combined Events

The NRC staff reviewed the applicant's evaluation of combined hydrologic events used to establish the PMF for the contributing drainage basins. The NRC staff evaluated whether the combinations of antecedent hydrologic and meteorological conditions are consistent with the guidance of ANSI/ANS-2.8-1992 for flooding from streams and rivers.

The NRC staff determined that the applicant based the combined-event analysis on Alternative I of ANSI/ANS-2.8-1992, which specifies the use of mean monthly baseflows, median antecedent soil moisture conditions, the controlling PMP event, and antecedent or subsequent precipitation equal to the lesser of (1) 40 percent of the controlling PMP or (2) the 500-year precipitation event. The NRC staff verified that this combination of hydrologic conditions was consistently applied throughout the applicant's PMF analyses for the Guadalupe River Basin, San Antonio River Basin, and West Coloma Creek watershed.

The NRC staff reviewed the applicant's justification for applying an antecedent storm rather than a subsequent storm. The applicant stated that the atmospheric conditions associated with a

PMP event would be expected to stabilize the atmosphere through subsidence following the storm, thereby making a subsequent precipitation event equal to 40 percent of the PMP meteorologically implausible. The applicant further placed the antecedent storm three days before the PMP event, which is within the three- to five-day interval recommended by ANSI/ANS-2.8-1992 and represents a meteorologically credible sequence for the region. The NRC staff finds that this approach is consistent with the guidance in ANSI/ANS-2.8-1992 and provides an acceptable basis for establishing antecedent watershed conditions for the PMF analyses.

The NRC staff also reviewed the applicant's comparison of the two antecedent precipitation alternatives permitted by ANSI/ANS-2.8-1992. The applicant compared the 40-percent PMP event with the 500-year precipitation event using the precipitation analyses presented in PSAR table 2.4.3-27 and determined that the 40-percent PMP produced the lesser precipitation volume. Accordingly, the applicant selected the 40-percent PMP antecedent storm for the combined-event simulations, consistent with the standard's requirement to use the lesser of the two alternatives.

Based on its review, the NRC staff finds that the applicant's combined-event methodology is consistent with the recommendations of ANSI/ANS-2.8-1992 for flooding from streams and rivers. The NRC staff concludes that the selected antecedent hydrologic conditions and storm sequencing provide an acceptable basis for development of the design-basis PMF used in the LMGS flood hazard evaluation.

2.4.3.1.3 West Coloma Creek Probable Maximum Flood

The NRC staff reviewed the applicant's evaluation of the PMF associated with the West Coloma Creek watershed, including the watershed characterization, PMP determination, hydrologic modeling, hydraulic modeling, coincident wind effects, and resulting flood levels at the proposed plant site. The NRC staff evaluated whether the methodologies, assumptions, and analytical tools used by the applicant are consistent with the guidance in ANSI/ANS-2.8-1992, Regulatory Guide 1.59, and accepted NRC guidance for design-basis flood evaluations.

Watershed Description

The NRC staff reviewed the description of the West Coloma Creek watershed for which the PMF model was set up. The PMF model included West Coloma Creek watershed, located in Calhoun County, Texas, using the 12-digit hydrologic unit code (HUC-12) watershed boundaries. These boundaries were refined near the LMGS site to delineate a direct tributary area, as shown in PSAR figures 2.4.3-27 through 2.4.3-29. The watershed extends from Powderhorn Lake in Matagorda Bay upstream about 4.5 mi (7.2 km) north of Green Lake, covering a total area of 102.32 mi² (265.01 km²). The drainage area upstream of the LMGS site totals 23 mi² (60 km²), of which 15.5 mi² (40.1 km²) flows directly toward the site; the remainder drains into East Coloma Creek. The Sparks watershed, located upstream of Sparks Road, encompasses 7.2 mi² (18.6 km²).

The NRC staff reviewed the applicant's characterization of the West Coloma Creek watershed and the delineation of the drainage area contributing to flooding at the proposed site. The applicant identified the watershed using the U.S. Geological Survey (USGS) 12-Digit Hydrologic Unit Code (HUC-12) boundaries and refined these boundaries to define the direct tributary area

relevant to the Texas Site. The watershed extends from Powderhorn Lake in Matagorda Bay to approximately 4.5 mi (7.2 km) north of Green Lake and encompasses a total drainage area of approximately 102.32 mi^2 (265.01 km^2). The NRC staff notes that the applicant identified a contributing drainage area of approximately 23 mi^2 (60 km^2) upstream of the LMGS site, of which approximately 15.5 mi^2 (40.1 km^2) drain directly toward the site. The remaining area contributes to the adjacent East Coloma Creek watershed.

The NRC staff reviewed the topographic data used in the analysis and finds that the applicant appropriately utilized the 2018 USGS LiDAR dataset for South Texas, which provides high-resolution terrain information suitable for hydrologic and hydraulic modeling. The dataset has a nominal pulse spacing of approximately 2.3 ft (0.8 m) and a reported vertical accuracy of approximately 8 in. (5.8 cm) at the 95 percent confidence level. The NRC staff also notes that the applicant used the LiDAR-derived terrain to evaluate critical flow paths and overtopping elevations along Farik Road and to characterize the local topography surrounding the LMGS site.

Based on its review, the NRC staff finds that the watershed delineation, contributing drainage areas, and topographic characterization provide an appropriate technical basis for subsequent PMF analyses.

Probable Maximum Precipitation

The NRC staff reviewed the applicant's development of PMP for the West Coloma Creek watershed. The applicant derived PMP values using the procedures contained in NWS Hydrometeorological Reports (HMRs) Nos. 51 and 52. The NRC staff finds that these documents represent acceptable guidance for estimating PMP for inland drainage basins.

The applicant developed the required DAD relationships using HMR No. 52 and conservatively applied the PMP developed for a 10-mi^2 (26-km^2) storm area across the entire watershed rather than developing separate PMP distributions for multiple storm centers. Because the drainage area directly contributing runoff to the LMGS site is approximately 15.5 m^2 (40.1 km^2), the NRC staff finds that application of the smaller-area PMP across the entire watershed is conservative.

The NRC staff further notes that intermediate 6-hour precipitation depths were developed by interpolation between the published HMR duration values to establish the precipitation input required for hydrologic modeling. Consistent with ANSI/ANS-2.8-1992, the applicant also incorporated an antecedent rainfall event equal to 40 percent of the 72-hour PMP occurring three days before the PMP event. The standardized 72-hour temporal rainfall distribution specified in HMR No. 52 was then used to develop the PMP hyetograph.

Based on its review, the NRC staff concludes that the applicant's PMP development is consistent with applicable regulatory guidance and provides a conservative precipitation input for PMF evaluation.

Precipitation Losses

The NRC staff reviewed the applicant's estimation of precipitation losses using the SCS curve number methodology. The NRC staff finds that the curve number method is an acceptable and widely used approach for estimating rainfall-runoff relationships.

The applicant determined land use characteristics using the National Land Cover Database and hydrologic soil classifications using the NRCS Soil Survey Geographic Database. The NRC staff notes that cultivated crops and pasture constitute the predominant land cover, while hydrologic soil group D, representing soils with relatively high runoff potential, predominates throughout the watershed.

Although the combination of land use and soil characteristics corresponds to a curve number of approximately 87, the applicant conservatively applied a curve number of 90 throughout the watershed. The NRC staff finds that this assumption increases predicted runoff volumes and therefore provides additional conservatism in estimating PMF runoff.

Accordingly, the NRC staff concludes that the applicant's precipitation loss assumptions are acceptable for design-basis flood analyses.

Runoff Hydrograph Determination

The NRC staff reviewed the applicant's development of runoff hydrographs for the West Coloma Creek watershed. The applicant modeled runoff from the upstream Sparks watershed using HEC-HMS Version 4.11 while representing the remainder of the watershed within the two-dimensional HEC-RAS hydraulic model through direct rainfall application.

The NRC staff notes that the applicant performed sensitivity analyses demonstrating overtopping of Farik Road during the PMP event and therefore appropriately represented upstream inflows through a combination of hydrologic and hydraulic models.

The applicant incorporated several conservative modeling assumptions, including:

- A curve number of 90;
- A basin lag time of 10 minutes, substantially shorter than the calculated lag time of approximately five hours;
- The PMP hyetograph developed in PSAR section 2.4.3.2.2; and
- An impervious area of 20 percent despite actual imperviousness being less than 5 percent.

The NRC staff finds that these assumptions produce increased runoff rates and earlier peak flows, thereby providing conservative boundary conditions for hydraulic modeling.

The resulting runoff hydrographs produced peak discharges of approximately 10,000 ft3/s (283 m^3/s) during the antecedent 40 percent PMP event and approximately 25,000 ft^3/s (708 m^3/s) during the full PMP event. The NRC staff concludes that these hydrographs provide an appropriate and conservative basis for the subsequent hydraulic analyses.

Hydraulic Model Setup and Flood Routing

The NRC staff reviewed the applicant's hydraulic evaluation of flooding using HEC-RAS Version 6.4.1 in two-dimensional (2D) mode. The NRC staff finds that use of a 2D hydraulic model is appropriate because it allows floodwaters to propagate naturally across the terrain rather than constraining flow to predefined channels.

The applicant constructed the terrain model using 2018 USGS LiDAR data and incorporated significant hydraulic controls, including roads, culverts, embankments, buildings, tanks, and drainage features. Model boundaries were established sufficiently distant from the nuclear island/conventional island (NI/CI) area to minimize boundary-condition effects on predicted flood elevations.

The NRC staff reviewed the model boundary conditions, including the upstream hydrograph from the Sparks watershed, downstream and western normal-depth boundaries, and direct rainfall applied over the model domain. The NRC staff finds these boundary conditions appropriate for simulating PMP-induced flooding.

The applicant utilized a computational mesh consisting of approximately 275,000 cells with local refinement around the LMGS site, drainage channels, and hydraulic structures. Sensitivity analyses performed by the applicant demonstrated that further mesh refinement did not materially affect model results. The NRC staff finds that the selected mesh resolution provides adequate numerical accuracy while maintaining computational efficiency.

The NRC staff also reviewed Manning's roughness coefficients assigned throughout the model using accepted engineering references and notes that conservative values were selected to maximize predicted flood depths.

The applicant evaluated the influence of roadway crossings and culverts and determined through sensitivity analyses that these hydraulic structures did not significantly affect predicted flooding because the PMP event produces extensive overbank flow across the relatively flat floodplain. The NRC staff finds this conclusion reasonable.

Model results indicate that maximum flood elevations adjacent to the LMGS site are approximately 31 ft (9.4 m) NAVD 88, with localized shallow inundation limited primarily to direct rainfall accumulation at the site perimeter. Peak flow velocities near the site are approximately 2 ft/s (0.6 m/s), which are below published erosion thresholds for appropriately protected surfaces. The applicant proposes erosion protection using properly graded rock armoring and geotextile where warranted.

Based on its review, the NRC staff concludes that the hydraulic modeling methods, assumptions, sensitivity analyses, and resulting flood elevations provide an acceptable basis for evaluating PMF impacts at the LMGS site.

Coincident Wind-Wave Activity

The NRC staff reviewed the applicant's evaluation of coincident wind effects on flood levels within the West Coloma Creek watershed. The applicant calculated wind-generated wave characteristics, including wind setup and wave height, using the same methodology previously applied for the Guadalupe River evaluation.

The applicant determined a design H1% wave height of approximately 1.19 ft (0.36 m). The NRC staff notes that West Coloma Creek is an ungauged drainage feature that remains dry during normal conditions and does not exhibit sustained base flow because it functions primarily as an agricultural drainage channel.

Based on its review, the NRC staff finds that omission of base flow from the coincident wind analysis is appropriate and that the applicant's wave calculations provide an acceptable basis for estimating coincident wind effects during the PMF event.

PMF Water Level Over West Coloma Creek

The NRC staff reviewed the applicant's determination of the PMF water level at the LMGS site, including the effects of coincident wave activity. Hydraulic model results indicate a maximum stillwater elevation of approximately 31 ft (9.4 m) NAVD 88 adjacent to the LMGS site.

The applicant calculated a maximum wave runup elevation of approximately 33.56 ft (10.23 m) NAVD 88. This value exceeds the proposed finished grade elevation of approximately 31.5 ft (9.6 m) NAVD 88 by approximately 2 ft (0.6 m).

The NRC staff notes that the calculated wave runup represents the maximum vertical extent of wave action on the embankment rather than sustained floodwater levels at the site. The applicant has incorporated this value into the site's design-basis flood evaluation to ensure that SSCs important to safety remain protected during the combined PMF and coincident wind event.

Based on its review of the hydrologic analyses, hydraulic modeling, and wave runup evaluation, the NRC staff finds that the applicant has established an acceptable design-basis PMF water level for West Coloma Creek consistent with applicable regulatory guidance.

2.4.3.2 Conclusions

Based on its review of the analyses and supporting documentation, the NRC staff finds that the predicted flood elevations and wave runup do not result in inundation of the LMGS site. The NRC staff also finds that the maximum wave runup occurs west of the site boundary and that intervening topographic features, transportation infrastructure, local drainage characteristics, and ground slope provide additional physical barriers that limit the potential for wave runup-related flooding to affect SR SSCs.

For the West Coloma Creek PMF evaluation, the applicant determined that the maximum stillwater elevation from the HEC-RAS 2D hydraulic analysis is approximately 31 ft (9.4 m) NAVD 88 at the LMGS site prior to consideration of wind-induced wave effects. The applicant calculated the total maximum water surface elevation including wave runup to be 33.56 ft (10.23 m) NAVD 88. This indicates a flood depth of about 2 ft (0.6 m) above the finished grade elevation of 31.5 ft (9.6 m) NAVD 88. The NRC staff reviewed the applicant's hydraulic analyses and the information provided during the audit regarding the planned site grading and surface drainage design. Although the final grading and surface water management features were not explicitly represented in the hydraulic model, the applicant indicated that these features are expected to provide positive drainage and prevent sustained accumulation of floodwaters in the vicinity of SR SSCs. The NRC staff acknowledges the applicant’s statement regarding the final site grading and the potential for additional margin and to address further refinements of the model at the OL stage. The site grading and passive engineered flooding protection design permit condition requires that LME notify the NRC staff once these aspects of the design and analysis are available.

The NRC staff also reviewed the applicant's PMF evaluation for the San Antonio Basin and the supporting watershed hydrology information contained in the InFRM Watershed Hydrology Assessment for the Guadalupe River Basin. Based on its review of these supporting analyses and the information provided during the audit process, the NRC staff finds that the applicant appropriately characterized the hydrologic conditions affecting the site and adequately evaluated the applicable PMF hazards associated with the contributing watersheds.

Based on its review of the PSAR, supporting technical reports, audit discussions, responses to audit questions, hydrologic and hydraulic analyses, hydraulic modeling results, coincident wind-wave evaluations, and proposed site protection features, the NRC staff concludes that the applicant has adequately evaluated the PMF hazards associated with the Guadalupe River, West Coloma Creek, and the San Antonio Basin.

2.4.4 Potential Dam Failures

PSAR section 2.4.4 evaluates the hydrological and seismic design basis for consideration of potential dam failures and failures of other water control and storage structures, thus resulting in flooding at the site.

The applicant evaluated potential flooding at the LMGS site resulting from failure of upstream dams in the Guadalupe River and San Antonio River Basins and from failure of onsite operating basins.

2.4.4.1 Technical Evaluation

For upstream dam failures, the applicant used the HEC-HMS and HEC-RAS models described in PSAR section 2.4.3 and applied the methodology described in NRC guidance JLD-ISG-2013-01, “Guidance for Assessment of Flooding Hazards due to Dam Failure,” Revision 0 (ML13151A153). The applicant identified potentially critical dams in the Guadalupe River Basin and San Antonio River Basin and determined that the Guadalupe River Basin represents the controlling upstream dam-failure hazard based on reservoir storage and other basin characteristics.

The applicant evaluated sunny-day, hydrologic, and seismic failure scenarios. For the principal bounding analysis, the applicant conservatively assumed failure of the potentially critical dams in conjunction with a 500-year rainfall event. Separate hydrologic and seismic analyses were performed for Canyon Lake Dam to demonstrate that the selected bounding condition adequately enveloped those failure mechanisms.

The applicant used the Froehlich (2008) methodology to estimate breach parameters and routed the resulting breach hydrographs through the HEC-HMS and HEC-RAS models. The applicant also evaluated wind setup and wave runup and considered the effects of intervening topographic and man-made features between the Guadalupe River and the LMGS site.

The applicant separately evaluated postulated failure of onsite operating Basins 5 and 31. Although the applicant concluded that failure of the basin embankments is not a credible event, the applicant conservatively assumed simultaneous failure for the flooding analysis. The evaluation considered conservative initial water levels, instantaneous breach formation, failure of internal dikes, runoff, evaporation, hydraulic roughness, and sedimentation.

The applicant used HEC-RAS 2D to evaluate propagation of the basin-breach flood across the site. The applicant reported maximum inundation depths at the LMGS site ranging from approximately 5 to 12 in (0.13 to 0.3 m) for the analyzed scenarios, with localized depths of approximately 1.3 ft (0.4 m) at portions of the site. Localized flood velocities were calculated to reach approximately 11 ft/s (3.4 m/s).

The applicant's current analysis does not incorporate final site grading and final site drainage and flood protection features. The applicant stated that final site grading and surface water management systems will provide additional protection and are expected to prevent inundation of SR SSCs.

2.4.4.1.1 Compilation of the Upstream Dam Database

The applicant identified upstream dams in the Guadalupe River and San Antonio River Basins using the National Inventory of Dams and information obtained from the TCEQ. The applicant identified 29 dams on the Guadalupe River and its tributaries and 35 dams on the San Antonio River and its tributaries with storage capacities greater than 3,000 acre-feet (3.7 million m^3) upstream of the LMGS site.

The applicant determined that the Guadalupe River Basin represents the controlling upstream dam-failure hazard based on the substantially greater storage capacity of Canyon Lake Dam and the greater aggregate storage of the potentially critical dams in the Guadalupe River Basin.

The NRC staff reviewed the applicant's dam inventory, selection criteria, storage comparisons, and supporting information during audits. The NRC staff finds that the applicant adequately identified the potentially significant upstream dams for the purpose of the CP stage flood hazard evaluation.

The NRC staff also finds the applicant's determination that the Guadalupe River Basin provides the controlling upstream dam-failure scenario reasonable based on the information provided. The determination is supported by the relative reservoir storage and the subsequent hydraulic analyses used to evaluate the potential flood effects at the LMGS site. Therefore, the NRC staff finds the information provided in this subsection acceptable.

2.4.4.1.2 Selection of Dams and Their Combinations for Failure Scenarios

The applicant evaluated potential combinations of dam failures using conservative deterministic assumptions consistent with the NRC Interim Staff Guidance (ISG), JLD-ISG-2013-01. The applicant considered sunny-day, hydrologic, and seismic dam-failure mechanisms and evaluated both individual system dams and groups of non-system dams represented by hypothetical dam clusters.

The NRC staff reviewed the applicant's selection process and the bases for the failure combinations. The NRC staff finds that the applicant's approach is consistent with the guidance in JLD-ISG-2013-01 and, therefore, provides reasonable assurance that potentially significant upstream dam-failure mechanisms have been considered.

Screening of Inconsequential and Noncritical Water Control Structures

The applicant identified 29 upstream dams within the Guadalupe River Basin and its tributaries with storage capacities greater than 3,000 acre-feet for evaluation in the dam-failure flood hazard assessment. The applicant provided the principal characteristics of these dams, including dam height, length, storage capacity, dam type, and year of completion, in PSAR table 2.4.4-2.

The applicant excluded dams with storage capacities less than 3,000 acre-feet from further consideration based on the conclusion that their failure would not produce a more limiting flood hazard than the evaluated scenarios. The applicant also stated that all 29 retained dams are classified as high-hazard dams in the U.S. Army Corps of Engineers (USACE) National Inventory of Dams and, consistent with NRC ISG, JLD-ISG-2013-01, considered them to be "potentially critical" dams for the site-specific dam-failure evaluation.

The applicant categorized the 29 potentially critical dams into two groups: (1) "System" dams located on the main stem of the Guadalupe River and (2) "Non-System" dams located on Guadalupe River tributaries. The applicant identified eight system dams; Dunlap, McQueeney, Placid, Meadow, Gonzales, Wood, Canyon, and Coleta for evaluation as part of the site-specific dam-failure analysis, with the remaining dams classified as non-system dams. The locations of the system dams are provided in PSAR figure 2.4.4-3.

Hypothetical Dam Clusters

The applicant evaluated the 22 non-system dams in the Guadalupe River Basin using the simplified hydrologic modeling approach described in section 3.2 of NRC ISG, JLD-ISG-2013-01. The applicant grouped hydraulically connected non-system dams into watershed-based clusters to reduce the number of breach simulations while maintaining conservative assumptions for the dam-failure analysis.

The applicant grouped non-system dams into three hypothetical dam clusters based on their hydraulic connectivity and watershed characteristics. The maximum pool volumes of the component dams were combined to establish conservative hypothetical reservoirs. Elevation-storage relationships were developed using topographic data, and the resulting hypothetical dams were incorporated into the hydrologic model.

The applicant developed three hypothetical dams representing the Comal River, Plum Creek, and York Creek/San Marcos River tributary watersheds. For each hypothetical dam, the applicant combined the maximum storage volumes of the hydraulically connected component dams and located the hypothetical dam at the downstream-most point of the cluster or at the confluence of the contributing tributaries. The applicant stated that hypothetical dams representing dams in series were modeled off-stream within the HEC-HMS to avoid artificial attenuation of upstream flood flows.

The applicant developed elevation-storage relationships for each hypothetical dam using a 10-meter digital elevation model (DEM). The applicant used these relationships to establish the initial reservoir water surface elevations corresponding to full pool conditions and determined the hypothetical dam heights from the DEM-derived invert elevations. The applicant provided cluster configurations, hypothetical dam locations, and elevation-storage relationships in PSAR

table 2.4.4-2 and PSAR figures 2.4.4-4 through 2.4.4-9 to support the site-specific dam-failure analysis.

The NRC staff therefore finds the applicant's evaluation under this subsection acceptable for the CP stage because it is consistent with the guidance in JLD-ISG-2013-01 (ML13151A153).

Dam Failure Permutations

The applicant evaluated combinations of dam failures intended to bound potential hydrologic, seismic, and sunny-day failure mechanisms. The NRC staff reviewed the applicant's selection of failure permutations and the associated assumptions during audit discussions with the applicant's subject matter experts.

The NRC staff finds that the selected permutations provide a conservative basis for determining the maximum upstream dam-failure flood level at the site.

Failures of Upstream Dams in the Guadalupe River Basin

The applicant identified Canyon Lake Dam and Coleto Creek Dam as particularly significant structures based on reservoir volume, dam height, and proximity to the LMGS site. The applicant evaluated the potential failure of the Guadalupe River Basin dams using the failure mechanisms specified in ISG, JLD-ISG-2013-01.

The NRC staff reviewed the applicant's treatment of these dams and the associated failure scenarios and finds the approach acceptable because it considers the most significant dam structures located close to the LMGS site and evaluated them following the guidance in ISG, JLD-ISG-2013-01.

Sunny-Day Failure Scenarios

The applicant postulated simultaneous failure of the potentially critical dams without a hydrologic or seismic initiating event and combined the failures with a 500-year rainfall event. The applicant used maximum pool conditions for the potentially critical reservoirs and a conservative sunny-day failure representation for Canyon Lake Dam.

The NRC staff reviewed the assumptions used to establish the sunny-day failure scenario and finds that the scenario provides a conservative assessment of the potential upstream dam-failure flood hazard. The NRC staff also reviewed the applicant's comparison of the sunny-day results with the hydrologic and seismic failure scenarios and finds that the applicant adequately demonstrated that the sunny-day scenario bounds the evaluated alternatives. Accordingly, the NRC staff finds the postulated sunny-day failure scenario described in this subsection acceptable.

Hydrologic Failure Scenarios

The applicant evaluated overtopping failure of Canyon Lake Dam and other potentially critical dams. For Canyon Lake Dam, the applicant performed an overtopping analysis using a top-of-dam pool elevation and a rainfall event. The resulting water surface elevation at the controlling river station was lower than that resulting from the applicant's bounding sunny-day and 500-year rainfall scenario.

The NRC staff reviewed the hydrologic failure methodology and results. The NRC staff finds that the analysis adequately demonstrates that the hydrologic failure scenario does not result in a more severe flood hazard at the site than the bounding scenario used by the applicant. The NRC staff therefore finds the evaluation provided in this section acceptable.

Seismic Collapse Failure Scenarios

The applicant considered the two seismic alternatives described in ISG, JLD-ISG-2013-01, and performed a conservative seismic breach evaluation for Canyon Lake Dam using a 500-year flood pool condition and an assumed rapid dam collapse.

The applicant's calculated peak water surface elevation was below the elevation resulting from the bounding sunny-day and rainfall scenario.

The NRC staff reviewed the seismic failure assumptions and results and finds that the applicant adequately demonstrated that the evaluated seismic collapse scenario is bounded by the selected design-basis dam-failure flood.

Accordingly, the NRC staff finds the postulated seismic failure scenario is acceptable because it follows the guidance in ISG, JLD-ISG-2013-01, and demonstrates that seismic dam breach is bounded by the scenario used by the applicant.

Failure of Onsite Water Control or Storage Structures

The applicant evaluated postulated failure of onsite operating Basins 5 and 31. Although the applicant concluded that failure of the basin embankments is not a credible event, the applicant conservatively assumed simultaneous failure of the basin embankments for purposes of evaluating the onsite flood hazard.

The NRC staff reviewed the basin configuration, potential failure mechanisms, initial pool conditions, breach assumptions, and resulting flood analyses. The NRC staff finds that the applicant appropriately considered failure of onsite water control or storage structures as a potential flooding mechanism and used conservative assumptions to characterize the resulting flood hazard.

The NRC staff notes, however, that the final site grading and final configuration of on-site drainage and passive flood protection features are not available at the CP stage. These features can affect the propagation, depth, velocity, duration, and recession of basin-breach floodwaters. As discussed in section 2.4.2.1 of this SE, the NRC proposed the site grading and passive engineered flooding protection design permit condition, which requires that LME notify the NRC staff once these aspects of the design and analysis are available.

2.4.4.1.3 Unsteady Flow Analysis of Potential Upstream Dam Failures

The applicant used HEC-HMS and HEC-RAS to simulate the hydrologic and hydraulic consequences of postulated upstream dam failures. The NRC staff reviewed the modeling methodology, model configuration, breach hydrographs, boundary conditions, and application of the models. For each evaluated dam-failure scenario, the applicant calculated dam-breach parameters for each potentially critical dam within the applicable HEC-HMS model component,

generated the corresponding breach hydrographs, and routed the resulting flood waves through the HEC-RAS unsteady-flow model to the LMGS site. The applicant evaluated the resulting Guadalupe River flood stages by comparing the calculated water surface elevations with the site and left bank topographic conditions to determine the potential for site inundation.

The NRC staff finds that the modeling approach followed by the applicant is appropriate for determining the propagation of dam-failure flood waves from the upstream reservoirs to the LMGS site.

Guadalupe River Basin Dam Failure Modeling

The applicant adapted existing HEC-HMS and HEC-RAS models for the Guadalupe River Basin to evaluate dam-failure scenarios. The NRC staff reviewed the modifications and supporting information provided by the applicant.

The NRC staff finds that the applicant's use of the existing basin models, with appropriate modifications to incorporate dam-breach hydrographs and the LMGS site, provides an acceptable basis for evaluating the upstream dam-failure flood hazard.

Physical Dam Data and Estimates of Breached Sections

The applicant used the Froehlich (2008) empirical methodology to estimate breach parameters, including breach width, side slopes, and breach formation time. The applicant used conservative breach assumptions for the system and hypothetical dams and a conservative sunny-day failure representation for Canyon Lake Dam.

The applicant used a 5-minute computational time step for the dam-breach simulations and applied breach side slope factors consistent with the Froehlich (2008) recommendations, including a side slope factor of 1.4 for overtopping failures and 0.9 for other failure modes.

For the system dams and hypothetical non-system dams, except Canyon Lake Dam, the applicant modeled dam failures using the HEC-HMS overtopping failure method. The applicant assumed an overtopping failure mode factor of 1.3 and a breach side slope ratio of 1 for these dams. For Canyon Lake Dam, the applicant modeled a conservative sunny-day failure scenario using a failure mode factor of 1 and a breach side slope ratio of 1. The resulting breach parameters were used to develop dam-failure hydrographs for routing through the hydraulic model.

The NRC staff reviewed the applicant’s dam-breach methodology, including the use of the Froehlich (2008) empirical methodology, the HEC-HMS overtopping failure method, the computational time step, and the assumed breach parameters. The NRC staff determined that the methods used are established approaches for estimating dam-breach parameters as described in JLD-ISG-2013-1 “Guidance for Assessment of Flooding Hazards Due to Dam Failure” and that the applicant’s breach assumptions are appropriately conservative for the CP stage evaluation. The NRC staff therefore determined that the dam-breach methodology and assumptions provide an acceptable technical basis for estimating dam-failure hydrographs and evaluating the resulting flood hazards at the LMGS site.

HEC-HMS Model Setup & Modifications

The applicant modified the HEC-HMS model to incorporate dam-breach hydrographs and the 500-year rainfall event. The applicant also adjusted routing capacities where combined dam-failure and rainfall flows exceeded the capacities of the original model rating curves.

The NRC staff reviewed the model modifications and the basis for the routing adjustments.

The original HEC-HMS model used a frequency-based elliptical design storm methodology to simulate basin-wide precipitation events. The applicant evaluated elliptical storm configurations using NOAA Atlas 14, Volume 11 precipitation data and performed a sensitivity analysis of seven gauge locations to identify the storm configuration producing the maximum basin response.

The applicant determined that the 500-year elliptical storm centered near the Cuero gauge produced the maximum peak discharge at the model outfall near Bloomington, Texas, with a peak flow of 773,141 ft^3/s (21,893 m^3/s). The applicant used this storm configuration as the controlling precipitation event for the dam-failure simulations. Dam-breach hydrographs were incorporated into the HEC-HMS model, and the timing of the breach hydrograph peaks was adjusted to coincide with the peak discharge timing of the selected 500-year elliptical storm at the respective dam-failure locations.

The applicant applied standard hydrologic modeling parameters, including NRCS deficit and constant loss methods, recession baseflow, Snyder Unit Hydrograph transformation, Lag, Modified Puls, and Muskingum routing methods. The applicant performed simulations over a 9-day period using a 15-minute computational time step. The HEC-HMS model upstream boundary was established at the Canyon Lake Dam outfall, and consistent with the USACE model configuration, Canyon Lake Dam outflows were disconnected from downstream reaches for the simulations. The model endpoint was the Guadalupe River near Bloomington, Texas, located approximately 15.5 mi (24.9 km) northwest of the LMGS site.

The applicant identified that the combined dam failure and 500-year elliptical storm scenarios produced peak flows exceeding the discharge capacities of certain original HEC-HMS reach rating curves. The applicant modified the routing parameters by increasing reach discharge capacities, primarily through adjustments to the Modified Puls routing parameters. Most reaches were increased by approximately 20 to 30 percent, with one reach representing Guadalupe River routing from the Canyon Lake outfall to New Braunfels increased by approximately 60 percent. The applicant used the resulting HEC-HMS peak discharge hydrographs as input to the HEC-RAS model for evaluation of downstream flood propagation and site impacts.

The NRC staff finds that the applicant appropriately addressed the increased flows associated with the combined dam-failure and rainfall scenarios.

HEC-RAS Model Setup & Modifications

The applicant used an existing HEC-RAS model of the Guadalupe River Basin and modified the model near the LMGS site to better represent site topography. The HEC-RAS model incorporated channel and floodplain geometry developed primarily from 2007–2008 Capital Area Council of Governments and Texas Natural Resources Information System LiDAR data. The applicant used 3-foot by 3-foot (0.9 m by 0.9 m) DEMs generated from the LiDAR data for

hydraulic modeling and supplemented areas beyond available cross sections with U.S. Geological Survey (USGS) 10-meter DEM data.

The applicant used the HEC-GeoRAS extension to develop and georeference cross sections and incorporated available field survey data, as-built drawings, and bridge and culvert information into the hydraulic model. Manning's roughness coefficients were developed using land use information, aerial imagery, and field observations. The applicant stated that omission of river bathymetric data results in conservative flood level estimates because the modeled channel elevations are higher and therefore provide reduced flow conveyance capacity.

For the dam-failure flood analysis, the applicant established the upstream boundary of the HEC-RAS model at the Canyon Lake Dam outfall (River Station 295.00) and the downstream boundary approximately 500 ft (152 m) northwest of Mission Lake in Calhoun County (River Station 0.00). The applicant selected River Station 1.255, located approximately 2,000 ft (610 m) northwest of the LMGS site as the downstream study location. The downstream boundary condition was defined using normal depth with a friction slope of 0.0001. The modeled Guadalupe River reach extends approximately 295 mi (475 km) and includes 1,211 cross sections.

The applicant modified River Station 1.255 to better represent the LMGS site and surrounding topography. The existing cross-section was extended approximately 12,845 ft (3,915 m) northwest from the original endpoint, following the vicinity of Jesse Rigby Road and extending beyond the West Coloma Creek intersection. The modified cross-section length was approximately 42,315 ft (12,898 m) and was used to evaluate flood conditions at the site.

The applicant incorporated eleven dam-breach hydrographs into the HEC-RAS model to represent the postulated failure scenarios throughout the Guadalupe River Basin. The breach hydrographs were grouped into two simulation cases, both evaluated concurrently with the 500-year frequency elliptical storm event. Case 1 included breach hydrographs for the system dams, including Canyon, Dunlap, McQueeney, Placid, Meadow, Gonzales, Wood, and Coleto dams. Case 2 included breach hydrographs for the hypothetical non-system dam clusters representing Cluster A, Cluster B, and Cluster C.

The applicant applied the dam-breach inflow hydrographs at selected upstream River Stations and maintained the remaining HEC-RAS inflow hydrographs consistent with the original USACE HEC-HMS model configuration. Of the 26 total inflow locations, 22 remained unchanged from the original model. Consistent with the original USACE model configuration, Canyon Lake Dam outflows were disconnected from downstream reaches in all simulations. The applicant also included the 500-year San Antonio River inflow at River Station 6.04. The HEC-HMS-generated inflow hydrographs were applied as boundary conditions in HEC-RAS, which was used to calculate the resulting water surface elevations at the LMGS site.

Wave Runup Analysis

The applicant evaluated wind setup, wave generation, wave runup, and potential overtopping associated with upstream dam failure flooding. The applicant considered multiple fetch alignments and combined the calculated wave effects with the dam-failure stillwater elevations.

The applicant concluded that potential wave runup exceedances occur west of the LMGS site and that intervening topographic and man-made features prevent the resulting water from

reaching the site. The applicant also evaluated potential overtopping and erosion of the river embankment.

The analysis was performed using the "Windspeed Adjustment and Wave Growth" module of the ACES within the CEDAS, Version 4.03. The applicant evaluated wind-generated wave conditions using site-specific fetch characteristics, including wind speed, wind duration, fetch length, average water depth, and fetch geometry.

The applicant evaluated 50 potential fetch alignments based on site topography and water depths obtained from the HEC-RAS dam-failure inundation results at the controlling river station. Fetches with water depths less than 1 ft (0.3 m) were excluded because wave generation would be limited by shallow water conditions. The applicant identified the controlling fetch alignment based on the maximum calculated significant wave height. Wind conditions were developed using the annual extreme-mile wind speed of 50 miles per hour (mph) (80 kilometers per hour (kph)) specified in ANSI/ANS-2.8-1992 and converted to applicable wave generation durations using the U.S. Army Corps of Engineers (USACE) Coastal Engineering Manual guidance.

The applicant calculated wind setup using U.S. Bureau of Reclamation (USBR) guidance and calculated wave runup using the USACE Coastal Engineering Manual methodology. Consistent with ANSI/ANS-2.8-1992 guidance, the applicant used the 1-percent wave height (H1%) as the design wave height for wave runup evaluations. The calculated wave setup and runup values were added to the dam-failure stillwater elevations to determine the maximum total water levels at the Guadalupe River left bank. The applicant reported calculated wave runup values ranging from approximately 0.5 to 5.85 ft (0.15 to 1.78 m) and wind setup values ranging from approximately 0.19 to 0.87 ft (0.06 to 0.27 m).

The applicant compared the calculated total water levels for the 50 fetch alignments with the left bank elevation of the Guadalupe River. The applicant concluded that areas where wave runup could exceed the riverbank elevation are located at least 1.5 mi (2.4 km) west of the LMGS site boundary. The applicant stated that intervening topographic and man-made features, including railroad crossings, basin areas, industrial facilities, local roads, and Highway 185, provide hydraulic controls that limit the progression of potential overtopping flows toward the site. In addition, the applicant identified a natural low-lying area west of the river and Highway 185 that would promote temporary storage and southward routing of potential overtopping flows rather than conveyance toward the LMGS site.

The applicant developed total water level inundation maps by combining the calculated wave effects with site topography and concluded that wave runup effects would not result in inundation of the LMGS site. The applicant further evaluated potential wave overtopping at the left bank of the Guadalupe River for the limiting fetch alignments and calculated overtopping volumes using the USACE Coastal Engineering Manual methodology. The applicant determined that the maximum potential overtopping volume was approximately 0.13 million gallons (0.49 million liters). The applicant evaluated this volume in combination with the controlling PMP event and concluded that downstream topographic features would prevent overtopping flows from reaching the site.

The applicant also evaluated the potential for wave-induced erosion of the Guadalupe River embankment using erosion threshold guidance from USACE, ERDC/CHL TR-10-7, "Flood-Side Wave Erosion of Earthen Levees: Present State of Knowledge and Assessment of Armoring

Necessity." The analysis considered wave height, embankment slope, wave duration, grass quality, and safety factors. The applicant calculated potential erosion depths ranging from approximately 0.01 to 0.74 ft (0.003 to 0.23 m), with an average value of approximately 0.12 ft (0.037 m), and concluded that wave-induced erosion would not compromise embankment integrity or result in a breach.

The NRC staff determined that the applicant's evaluation appropriately considered the relevant wave effects associated with upstream dam-failure flooding and that the use of the controlling fetch and design wave conditions provides a reasonable basis for evaluating the maximum potential wave effects. The NRC staff also determined that the calculated wave runup and overtopping results, together with the identified hydraulic controls and intervening terrain, demonstrate that potential overtopping flows would not propagate to the LMGS site. In addition, the NRC staff reviewed the applicant's evaluation of wave-induced embankment erosion and determined that the calculated erosion depths would not compromise the integrity of the river embankment.

Based on these reviews and the absence of a credible hydraulic pathway for the evaluated wave runup and overtopping effects to reach the site, the NRC staff determined that the applicant's analysis provides reasonable assurance that upstream dam-failure wave effects would not result in flooding of SR SSCs at the LMGS site and is acceptable because it follows established guidelines and methodologies developed by Federal agencies, including the U.S. Bureau of Reclamation (USBR) and the U.S. Army Corps of Engineers (USACE).

2.4.4.1.4 Failure of On-Site Water Control or Storage Structures Analysis

The applicant evaluated the potential flooding resulting from postulated breaches of the on-site operating basins (Basins 5 and 31) using the HEC-RAS two-dimensional (2D) Version 6.4.1 model. HEC-RAS 2D solves the two-dimensional shallow water equations or diffusion wave equations using an implicit finite volume numerical scheme. The applicant stated that the model formulation provides improved stability and allows larger computational time steps compared with traditional explicit numerical methods. The applicant also noted that HEC-RAS 2D uses a high-resolution sub-grid approach that incorporates detailed terrain information to develop hydraulic properties for computational cells and cell faces. The software is approved by the FEMA for use in Flood Insurance Studies.

The applicant used HEC-RAS 2D to simulate flood depths and flow rates resulting from postulated embankment breaches of Basins 5 and 31. The model incorporated site features that could affect flood propagation, including buildings, embankments, and other obstructions. The model domain was established to include the entire basin areas and surrounding areas near the LMGS NI and SR and NSRST SSCs. The applicant selected model boundaries through sensitivity analyses to ensure that boundary conditions did not influence calculated flood levels within the site area. The model domain generally followed the West and East Coloma Creek watershed boundaries, with the northern and southern extents limited to improve computational efficiency. The eastern boundary was established at the natural drainage outlet to the VBC.

The HEC-RAS 2D model encompassed approximately 18,000 acres (72.7 km^2) and extended approximately 4.5 mi (7.2 km) in the east-west direction and 7.5 mi (12 km) in the north-south direction. The applicant developed the computational mesh using the RAS Mapper module, with a base grid resolution of 40 ft by 40 ft (12 m by 12 m) and local refinement to 10 ft by 10 ft (3 m

by 3 m) in areas of interest, including West Coloma Creek, embankment boundaries, and site drainage features. The final model included approximately 689,222 computational cells. The applicant performed a grid sensitivity evaluation using a coarser 100-ft by 100-ft (30 m by 30 m) grid and concluded that the resulting water surface elevations were not significantly different from those calculated using the refined grid.

The applicant modeled site buildings as blocked features by representing them as elevated dry points at least 10 ft (3 m) above finished grade. This approach simulated the presence of building obstructions during basin-breach flooding and was used in the evaluation of flood levels at the LMGS site.

The NRC staff reviewed the applicant's HEC-RAS 2D model, including the model domain, computational grid, topographic data, boundary conditions, breach hydrographs, roughness coefficients, runoff assumptions, representation of site buildings and other obstructions, and sensitivity analyses. The NRC staff identified that the model domain and boundary conditions adequately encompass the areas that could affect flood propagation to the LMGS NI and SR SSCs. The NRC staff also determined that the computational grid provides sufficient spatial resolution to represent the relevant site features and drainage pathways and that the grid and boundary sensitivity evaluations demonstrate that the calculated flood levels at the site are not materially affected by the selected model configuration.

The NRC staff noted that HEC-RAS 2D is an established hydraulic modeling methodology and that the applicant appropriately incorporated site-specific topography and hydraulic features into the model. The representation of buildings as blocked features appropriately accounts for their potential obstruction of flood flows. Based on its review of the modeling methodology, input parameters, model configuration, and sensitivity evaluations, the NRC staff determined that the applicant's HEC-RAS 2D model provides an appropriate and sufficiently supported technical basis for evaluating the potential effects of on-site basin failures.

Assumptions in the On-Site Basin Breach Analysis

The applicant developed the on-site basin-breach flood analysis using conservative assumptions to evaluate the potential impacts of postulated failures of Basins 5 and 31. The applicant assumed no baseflow contribution from West and East Coloma Creeks because these channels are not continuously flowing and primarily function as stormwater conveyance features for upland drainage areas. The applicant also assumed no evaporation during the analysis to conservatively maximize the duration and depth of site inundation.

The applicant modeled buildings within the watershed as solid, impervious obstructions such that floodwaters would route around the structures rather than enter or pass through them. Security features, including delay barriers and Jersey barriers, were not included in the analysis due to the preliminary state of the design at this stage. The applicant developed site-specific SCS curve numbers based on soil characteristics, vegetation, and land use and applied conservative but credible values for watershed runoff calculations.

The applicant assumed that basin surface area remains constant with depth, resulting in the maximum available basin storage volume. The basin failure scenarios were developed using an event-based approach consistent with FEMA guidance, considering credible combinations of adverse basin failure mechanisms and initial pool conditions. Because emergency spillway

information was not available, the applicant conservatively assumed that basin water levels could rise to the top of the embankments and overflow during hydrologic events.

For the breach analysis, the applicant assumed instantaneous failure and removal of the breached embankment sections. The applicant also assumed that all internal basin dikes would fail concurrently with the main embankment breach and be removed instantaneously, resulting in the maximum potential release of stored basin water.

The NRC staff reviewed these assumptions and finds that they provide a conservative representation of on-site basin-breach flooding appropriate for the CP stage.

Topography and Bathymetry

The applicant used 2018 U.S. Geological Survey (USGS) LiDAR data for South Texas to develop the topographic representation for the LMGS site basin-breach flood analysis. The LiDAR dataset has a nominal pulse spacing of 2.3 ft (0.7 m) and a vertical accuracy of approximately 8 in. (0.2 m) at a 95 percent confidence level. The data are referenced to the North American Datum of 1983 Universal Transverse Mercator (UTM) Zone 14 horizontal coordinate system and the North American Vertical Datum of 1988 (NAVD 88), GEOID12B vertical datum.

The applicant incorporated the LiDAR-based topography into the HEC-RAS 2D model to represent the site area containing SR and NSRST SSCs. The applicant established the finished site grade elevation at 31 ft (9.4 m) NAVD 88 for the NI area.

The NRC staff reviewed the source and treatment of the topographic information.

The NRC staff finds the topographic information adequate for the CP stage analysis. However, the site grading design was not provided at the CP stage and may impact this analysis. As discussed in section 2.4.2.1 of this SE, the NRC staff addressed this by proposing a site grading and passive engineered flooding protection design permit condition, which requires that LME notify the NRC staff once these aspects of the design and analysis are available.

Boundary Conditions

The applicant established the HEC-RAS 2D model domain using four external boundaries. The applicant selected the northern and eastern boundaries sufficiently distant from the site such that the peak flood conditions at the LMGS SR and NSRST SSCs would occur before the flood wave reached these boundaries, thereby minimizing boundary effects on the calculated site flood levels. The applicant defined the southern and western boundaries using normal flow boundary conditions with a channel slope of 0.001.

The applicant established model boundaries sufficiently distant from the NI such that boundary effects would not control the predicted flood levels at SR SSCs. The applicant also performed sensitivity evaluations to establish the model extents.

The NRC staff reviewed the applicant’s model boundaries and boundary conditions and determined that the approach is consistent with the HEC-RAS 2D User’s Manual guidance for selecting model extents and boundary conditions to minimize their influence on areas of interest. The NRC staff also reviewed the applicant’s sensitivity evaluations and determined that

the boundaries are sufficiently distant from the NI and SR SSCs and do not materially affect the calculated peak flood levels at the site. Therefore, the NRC staff determined that the applicant's model boundaries and boundary conditions are acceptable for the CP stage evaluation.

Initial Conditions and Basin Failure Hydrographs

The applicant established initial reservoir conditions for the on-site basin-breach analysis consistent with ANSI/ANS-2.8-1992 guidance. The guidance recommends that the initial reservoir water level corresponds to one-half of the local PMP. The applicant identified the normal operating water level of the basins as 34.5 ft (10.5 m) NAVD 88 based on site drawings. The applicant calculated the LIP event as 55.7 in. (1.4 m) over 72 hours, corresponding to a water surface elevation of approximately 36.8 ft (11.2 m) NAVD 88, which is approximately 1.2 feet (0.37 m) below the top of the basin embankments.

For the hydrologic failure scenario, the applicant conservatively assumed that the basin water level could rise to the top of the embankment and used an initial water surface elevation of 37 feet (11 m) NAVD 88. For the sunny-day failure scenario, the applicant used the top-of-embankment elevation as the initial reservoir condition. The applicant set all initial flow velocities within the HEC-RAS 2D model domain to zero.

The applicant used the Froehlich (2008) methodology to estimate dam-breach peak discharge and breach formation time. Because the methodology does not define the complete breach outflow hydrograph, the applicant developed triangular breach hydrographs for each evaluated failure scenario using the calculated peak discharge and formation time parameters. The applicant assumed instantaneous removal of the breached embankment section and internal dikes.

The NRC staff reviewed the initial conditions and breach hydrograph assumptions and finds that the assumptions provide a conservative basis for determining on-site flood levels.

Model Parameter Selection

The applicant assigned Manning's roughness coefficients to each computational cell in the HEC-RAS 2D model based on land cover categories. The applicant considered published guidance from the U.S. Department of Agriculture (USDA) NRCS Technical Release 55 (TR-55), the HEC-RAS Reference Manual, and Chow's *Open-Channel Hydraulics* to establish appropriate ranges of roughness coefficients for overland floodplain flow.

The applicant stated that, because floodwaters are generally expected to route away from the NI area, higher Manning's roughness values would result in more conservative predictions of flood elevations at the site. Based on the conterminous United States land cover dataset, the model domain is primarily composed of cultivated crops and hay/pasture areas. The applicant therefore applied a Manning's roughness coefficient of 0.05 throughout the model domain.

The applicant calculated runoff characteristics using the SCS curve number method, which determines runoff response based on land use and hydrologic soil group. The applicant identified agricultural land use and hydrologic soil group D as the dominant watershed conditions, corresponding to a curve number of 87 based on USDA NRCS guidance. The applicant conservatively applied a curve number of 90 for the applicable watershed areas and

assigned a curve number of 99 within the site area to account for paved surfaces and reduced infiltration.

The NRC staff reviewed the applicant's selection of Manning's roughness coefficients and SCS curve numbers against the referenced USDA NRCS TR-55, HEC-RAS Reference Manual, and Chow's *Open-Channel Hydraulics*. The NRC staff determined that the selected Manning's roughness coefficient of 0.05 and curve numbers of 90 and 99 are appropriately supported by the site land cover, soil, and drainage characteristics and are conservatively applied for the flood analysis. Therefore, the NRC staff determined that the applicant's hydraulic roughness and runoff parameters are appropriately selected and provide an acceptable basis for evaluating flood levels at the LMGS site. The NRC staff reviewed the parameter selection and sensitivity evaluations and finds that the applicant's modeling parameters are appropriately selected based on the site conditions, and that the sensitivity evaluations provide reasonable assurance that the predicted flood levels are not unduly dependent on the computational grid or sedimentation assumptions.

Flood Levels from Failure of On-Site Water Control or Storage Structures

The applicant evaluated the potential flooding impacts from postulated breaches of the on-site operating basins considering the relative elevations of the basin embankments and the NI area. The finished floor elevation of the NI is approximately 31.5 ft (9.6 m) North American Vertical Datum of 1988 (NAVD 88), which is approximately 6.5 feet (2 m) below the crest elevation of Basins 5 and 31. The applicant evaluated two basin-breach scenarios with different breach locations along the northern embankment of Basin 5.

The applicant's results indicate maximum on-site inundation at the NI of approximately 32.5 ft (9.9 m) NAVD 88, corresponding to approximately 1 ft (0.3 m) above the reported NI finished floor elevation. The applicant also calculated localized velocities substantially higher than the general site velocities.

The applicant evaluated the potential effects of scour and sediment deposition resulting from high-velocity flows associated with basin-breach flooding. The applicant reviewed LMGS site topography and drainage features to determine potential sediment deposition areas. The applicant concluded that sediment deposition within West Coloma Creek is not credible because the creek is bounded by dikes that would prevent eroded material from entering the channel. The applicant also concluded that sediment deposition within LMGS site drainage ditches and areas surrounding the basins and LMGS site is unlikely because of the relatively high flow velocities in those areas. The applicant determined that potential sediment deposition would more likely occur downstream of the site in areas where flow velocities decrease.

For the sedimentation sensitivity evaluation, the applicant identified areas where flow velocities were below 1 ft (0.3 m) per second as potential deposition regions. The applicant conservatively increased the ground elevation of the identified sedimentation area, located approximately 2,000 ft (610 m) south of the site on the west side of the Coloma Creek floodplain, by 2 ft (0.6 m) to represent complete blockage due to sediment deposition. The applicant performed both basin-breach scenarios using the modified topography and concluded that sediment deposition downstream of the LMGS site does not significantly affect inundation depths at the LMGS site.

The applicant reported that, for scenario 1, maximum inundation depths at the LMGS site range from approximately 5 to 10 in (0.13 to 0.25 m), with localized flooding at the western site boundary reaching approximately 1 ft (0.3 m). For scenario 2, maximum inundation depths at the LMGS site range from approximately 6 to 12 in (0.15 to 0.3 m), with localized flooding at the western site boundary reaching approximately 1.3 ft (0.4 m). The applicant noted that scenario 2 primarily affects the southern portion of the LMGS site, whereas scenario 1 affects both the southern and northern portions of the LMGS site.

The applicant evaluated flood velocities resulting from the basin-breach scenarios and calculated velocities at the LMGS site of approximately 6 ft/s (1.8 m/s), with localized velocities near the LMGS site boundaries reaching approximately 10 ft/s (3 m/s). The applicant concluded that these velocities should be considered in the design of site berm protection features. The applicant also identified roads subject to greater than 1 ft (0.3 m) of inundation for both evaluated scenarios.

Based on the HEC-RAS 2D modeling results and sedimentation sensitivity evaluation, the applicant concluded that potential operating basin failures would not result in significant sediment deposition effects or increased flooding impacts at the LMGS site.

The applicant concluded that final site grading and stormwater management features, which were not included in the current analysis, would drain the predicted shallow flooding away from SR and NSRST SSCs.

The NRC staff considers the predicted flood levels and velocities significant to the final determination of flood protection adequacy. The site grading, drainage features, and passive flood protection features are not established at the CP stage. As discussed in section 2.4.2.1 of this SE, the NRC staff addressed this by proposing a site grading and passive engineered flooding protection design permit condition, which requires that LME notify the NRC staff once these aspects of the design and analysis are available. Detailed design information on site grading and drainage will be provided at the OL stage.

The NRC staff finds that the CP stage analysis adequately characterizes the potential hazard and provides sufficient information to support the issuance of a CP.

Flood propagation and the flood levels resulting from the postulated failure of on-site water storage structures could be impacted by changes to the site grade and inclusion of passive flood protection structures, which is addressed by the permit condition discussed above.

2.4.4.1.5 Water Levels at the Plant Site

The applicant's evaluations of upstream dam-failure scenarios within the Guadalupe River Basin and postulated failures of the on-site basin embankments identified would not result in the design-basis flood for the LMGS site. The applicant determined that from the postulated failure of upstream dams and on-site water control structures, the failure of the on-site basin embankments resulted in the limiting flood condition at the LMGS site.

Water Level at the Plant Site from the Failures of Upstream Dams

The applicant concluded that upstream dam failure does not result in flooding of the LMGS site because the maximum combined stillwater, wind setup, and wave runup elevations remain below the intervening riverbank and topographic features.

The NRC staff reviewed the calculated water surface elevations and wave effects and finds that the applicant adequately demonstrated that upstream dam failures do not result in flooding of the at the LMGS site and therefore finds this subsection acceptable for the CP stage.

Water Level at the Plant Site from the Failure of On-Site Water Control or Storage Structures

The applicant evaluated the potential flooding impacts at the NI and LMGS site resulting from postulated breaches of the northern embankments of the on-site operating basins. The applicant calculated a maximum flood water surface elevation of 32.5 ft (9.91 m) NAVD 88 at the NI and site area. The applicant did not include wind setup or wave runup effects in this evaluation because of the short duration of the basin-breach flooding event.

Based on the calculated flood elevation and the NI finished grade elevation, the applicant determined that SR and certain NSRST SSCs could experience a maximum inundation depth of approximately 1 ft (0.3 m) during an on-site basin failure scenario. The applicant stated that the final site grading and surface water management features, which were not included in the current hydraulic model, are expected to provide additional drainage and conveyance capacity to remove localized flooding and prevent inundation of SR and certain NSRST SSCs.

The NRC staff reviewed the calculated water level and the relationship between the predicted water surface elevation and the preliminary NI grade.

Because the site grading and drainage configuration are not available at the CP stage, as discussed in section 2.4.2.1 of this SE, the NRC staff proposed a site grading and passive engineered flooding protection design permit condition, which requires that LME notify the NRC staff once these aspects of the design and analysis are available. With the permit condition the NRC staff finds this analysis to be acceptable.

2.4.4.1.6 Dynamic Effects of Breach Flood Waves on Structures, Systems, and Components

The applicant identified the need for flood protection features for SR and certain NSRST SSCs, as discussed further in PSAR section 2.4.10. Based on the dam-failure evaluations performed for upstream dams and on-site operating basin embankments, the applicant determined that the maximum site inundation depth would be approximately 1 ft (0.3 m) for the evaluated dam-failure scenarios.

In PSAR section 2.4.4.6, the applicant states, in part, that with implementation of the final site grading and stormwater management systems, flooding associated with dam-failure scenarios is not expected to subject SR SSCs to adverse dynamic flood forces. The applicant further stated that the plant intake and outlet structures will be designed to withstand the anticipated dynamic loading conditions associated with the evaluated flooding scenarios.

The applicant evaluated flood velocities and concluded that the general flow velocities near the NI would be relatively low, although localized velocities could be significantly higher along the LMGS site periphery.

The NRC staff reviewed the applicant's evaluation of dynamic flooding effects and finds that the analysis performed for the CP stage adequately identifies the potential for localized high-velocity flow and associated erosion/scour effects. The NRC staff further finds that the applicant has appropriately recognized the need for protection of the NI perimeter.

The final evaluation of hydrodynamic loading, erosion, scour, and other dynamic effects should be performed using the final site configuration and flood protection features. Because of the potential impact of site grading design and the passive engineered flooding protection design on this analysis and the limited design information available in the PSAR on these design aspects, the NRC staff proposed a site grading and passive engineered flooding protection design permit condition, which requires that LME notify the NRC staff once these aspects of the design and analysis are available.

Potential for Landslide and Waterborne Missiles

The applicant evaluated the potential for secondary effects associated with flooding hazards, including landslide-induced river blockage and waterborne missiles. The applicant concluded that a major landslide resulting in blockage of the Guadalupe River near the LMGS site is highly unlikely due to the broad, relatively flat floodplain and the mild slopes of the surrounding river valley.

The applicant also concluded that upstream dam-failure flooding would not transport waterborne missiles to the LMGS site. For on-site basin failure, the applicant acknowledged the potential for debris transport but concluded that the associated debris effects would not be significant compared with other external hazards.

The applicant determined that waterborne missile effects from upstream dam failures are not applicable because the LMGS site is located above the Guadalupe River floodplain and upstream dam-failure scenarios are not expected to result in flood flows reaching the LMGS site.

The applicant acknowledged the potential for waterborne debris associated with postulated breaches of the on-site operating basin embankments. However, the applicant concluded that flood wave velocities would attenuate before reaching SR and certain NSRST SSCs within the NI area. The applicant further concluded that debris transported by basin-breach floodwaters would not represent a critical design consideration for SR and certain NSRST SSCs when compared with the effects of tornado-generated missiles.

The NRC staff reviewed the applicant's evaluation and finds that the potential for landslide-induced blockage and waterborne missiles associated with upstream dam failure does not represent a significant flooding hazard to the LMGS site. The NRC staff also finds that the potential for debris generated by on-site basin failure is appropriately recognized, however the evaluation in the application did not consider potential impacts from site grading or passive engineered flooding protection features, for which design was unavailable at the time of CP. As discussed in section 2.4.2.1 of this SE, the NRC staff proposed a site grading and passive

engineered flooding protection design permit condition, which requires that LME notify the NRC staff once these aspects of the design and analysis are available.

2.4.4.2 Conclusions

The NRC staff reviewed the applicant's evaluation of potential flooding resulting from upstream dam failures and failure of on-site water control or storage structures. The NRC staff's review included review of reports, analyses, model documentation, and supporting information provided through the regulatory audit process.

Based on its review of the information provided in PSAR section 2.4.4, including the applicant's dam-failure analyses, hydrologic and hydraulic modeling, wave runup evaluation, on-site basin-breach analysis, and responses to audit questions, the NRC staff finds that the applicant provided sufficient information to evaluate the potential dam-failure flooding hazards at the LMGS site. The NRC staff determined that the analyses adequately characterize the magnitude and effects of the evaluated flood hazards.

Due to the preliminary state of the design, final site grading and associated drainage and passive flood protection features are not established at the CP stage. The NRC staff concludes that the application with the CP condition adequately addressed the potential dam-failure flooding hazard for purposes of the construction permit application and that there is reasonable assurance that the facility can be constructed at the proposed location without undue risk to the health and safety of the public.

2.4.5 Probable Maximum Surge and Seiche Flooding

PSAR section 2.4.5 describes flooding resulting from probable maximum surge and seiche flooding at the LMGS site. The NRC staff review assesses whether the application provides adequate information and supports technical analyses describing the hydrometeorological basis, along with the probable maximum surge and seiche evaluations, developed using site-specific topographic, bathymetric, and hydrologic characteristics incorporated into the model. The NRC staff reviewed this portion of the application to determine if the applicant described the hydrometeorological design basis used to determine the extent of any flood protection required for SR and certain NSRST SSCs.

2.4.5.1 Technical Evaluation

The NRC staff reviewed the applicant's evaluation of probable maximum surge and seiche flooding for the LMGS site. The NRC staff reviewed reports, calculations, modeling documentation, and supporting information provided during the regulatory audit.

The applicant first performed a screening-level evaluation using the NOAA Sea, Lake, and Overland Surges from Hurricanes (SLOSH; NOAA, 1992, 2017, and 2021) model to identify limiting hurricane scenarios. The applicant then performed refinement-level analyses using the ADvanced CIRCulation (ADCIRC, 1994) model coupled with the Simulating WAves Nearshore (SWAN) model to evaluate the limiting storm scenarios identified during the screening analysis.

The NRC staff evaluated the applicant's methodology for determining the probable maximum storm surge (PMSS), including the selection of probable maximum hurricane (PMH) parameters, development of synthetic hurricane scenarios, calculation of the antecedent water

level, screening-level SLOSH analyses, refined ADCIRC+SWAN analyses, model calibration and verification, and evaluation of wave effects and wave runup.

The NRC staff's review was performed against the applicable acceptance criteria in section 2.4.5 of NUREG-0800 and the applicable regulatory requirements for protection of SSCs important to safety from natural phenomena. PDC 2 requires SSCs important to safety to withstand the effects of natural phenomena, including hurricanes, floods, and seiches, without loss of capability to perform their safety functions. The NRC staff also considered the design-basis flood estimation principles described in NUREG/CR-7046, including the use of progressively refined, site-specific hydrologic and hydrodynamic analyses.

2.4.5.1.1 Probable Maximum Winds and Associated Meteorological Parameters

The applicant identified the PMH as the controlling meteorological event for development of the PMSS at the LMGS site. The applicant stated that, consistent with the guidance in NUREG-0800, section 2.4.5, the PMH is the controlling event for Gulf Coast sites and that neither a probable maximum windstorm nor a moving squall line would produce the PMSS at the LMGS site. The applicant developed the PMH wind field using the meteorological parameters provided in NWS Report 23, including the peripheral pressure (pp), central pressure (pc), radius of maximum winds (Rm), forward speed (T), and track direction (θ) for the shoreline location nearest the LMGS site.

The applicant compared the PMH parameters with the historical hurricane record for the Texas Gulf Coast, including the 1886 Indianola Hurricane, Hurricanes Carla, Katrina, Rita, and Harvey, and concluded that the NWS Report 23 PMH parameters remain sufficiently conservative for establishing the design-basis hurricane despite more recent hurricane activity.

The NRC staff reviewed the applicant's methodology for selecting the controlling meteorological event and developing the PMH parameters. The NRC staff finds the applicant's methods and processes acceptable because they are consistent with the guidance in NUREG-0800, section 2.4.5, and utilize the PMH methodology documented in NWS Report 23, which is an NRC-accepted basis for defining probable maximum hurricane characteristics. The NRC staff also finds the applicant's evaluation of historical hurricane data acceptable because it provides reasonable confirmation that the selected PMH parameters conservatively bound historically observed storms affecting the site region.

The NRC staff reviewed the basis for selection of the PMH parameters and the range of synthetic storm characteristics. The NRC staff also reviewed the applicant's adjustment of the central pressure deficit in the SLOSH simulations to ensure that the modeled wind speeds remained within the applicable PMH wind speed limits.

The NRC staff determined that the applicant considered the meteorological parameters that have the potential to maximize storm surge at the LMGS site and used a systematic approach to identify the storm characteristics requiring refined analysis. Therefore, the NRC staff finds the applicant’s methodology acceptable for the CP stage.

The NRC staff further notes that the PMH parameters used in the analysis were derived from NWS 23 (NOAA, 1979) and that the applicant considered historical hurricanes affecting the Texas Gulf Coast. The applicant's comparison of the PMH central pressure with the historical

Indianola Hurricane provides additional support for the conservatism of the selected PMH parameters.

Therefore, the NRC staff concludes that the applicant's evaluation of probable maximum winds and associated meteorological parameters is acceptable for the CP stage.

2.4.5.1.2 Surge and Seiche Water Level

The applicant described the LMGS site as being located approximately 16 mi (26 km) upstream of the Gulf Coast on Coloma Creek within the San Antonio Bay and Matagorda Bay system.

The applicant evaluated the propagation of PMSS through the Gulf of America (formerly known as Gulf of Mexico), Matagorda Bay, San Antonio Bay, and the Coloma Creek system to the LMGS site. The applicant used a hierarchical approach consisting of a screening-level SLOSH analysis followed by a refined ADCIRC+SWAN analysis.

The applicant's approach considered the site-specific bathymetry and topography, storm characteristics, antecedent water level, storm surge propagation, wave effects, and wave runup. The NRC staff reviewed the applicant's approach and determined that the use of progressively refined hydrodynamic models is appropriate for characterizing the PMSS hazard at the site. The implementation of the progressive refinement is consistent with the hierarchical hazard assessment approach described in NUREG/CR-7046.

The applicant's analysis determined that seiche effects would not materially contribute to flooding at the LMGS site. The applicant also evaluated historical hurricane events and used Hurricane Harvey and Hurricane Carla for model verification and comparison with observed or reported water levels.

The NRC staff finds that the applicant's analysis appropriately considers both the meteorological forcing and the regional hydraulic pathways by which storm surge can propagate to the LMGS site.

The NRC staff reviewed the applicant's characterization of the LMGS site and the potential flood mechanisms. The NRC staff finds the applicant's methods and processes acceptable because they appropriately identify storm surge as the controlling coastal flood mechanism and provide an adequate technical basis for concluding that seiche effects are not a credible flooding hazard for the LMGS site. The NRC staff further finds that the applicant appropriately identified the hydraulic pathways through which coastal storm surge could propagate to the LMGS site.

Historical Hurricane Events and Storm Surge

The applicant evaluated historical hurricanes that made landfall along the Texas Gulf Coast between 1851 and 2023 and identified the 1886 Indianola Hurricane as the most intense historical storm affecting the vicinity of the LMGS site. The applicant also evaluated Hurricane Carla (1961) and Hurricane Harvey (2017), including documented storm surge elevations and observed inundation in the surrounding coastal region. The applicant concluded that hurricanes making landfall outside the immediate vicinity of the site would have comparatively small effects on storm surge at the LMGS site.

The NRC staff reviewed the applicant's compilation and evaluation of historical hurricane and storm surge information. The NRC staff finds the applicant's methods and processes acceptable because the historical record provides an appropriate basis for understanding regional storm surge behavior and supports the applicant's selection of the PMH as the controlling design-basis event. The NRC staff also finds that the historical information provides appropriate context for evaluating the conservatism of the applicant's PMSS analysis.

The NRC staff reviewed the historical event selection and the associated model comparisons and finds that the historical hurricane evaluation provides an appropriate basis for assessing model performance and regional storm surge behavior. Therefore, the NRC staff concludes that the applicant's evaluation of historical hurricane events and storm surge is acceptable for the CP stage.

Storm Surge Analysis

The applicant estimated the PMSS at the LMGS site using a hierarchical hazard assessment methodology consistent with NUREG/CR-7046. The applicant used a two-phase modeling approach consisting of SLOSH screening analyses and refined ADCIRC+SWAN analyses. The NRC staff reviewed the principal modeling assumptions, input parameters, model domains, model resolution, model verification and calibration, and production-run results.

The applicant calculated an antecedent water level of 4.0 ft (1.2 m) NAVD 88 based on the 10-percent exceedance high tide and projected sea-level rise.

The NRC staff reviewed the principal modeling assumptions, input parameters, model domains, model resolution, model verification and calibration, and production-run results. The NRC staff determined that consideration of the antecedent water level and long-term sea-level rise is appropriate because these factors represent the water level conditions that can reasonably be present before the occurrence of the postulated maximum storm surge and are consistent with the treatment of antecedent water levels and future site conditions in NUREG/CR-7046. The NRC staff also determined that incorporating projected sea-level rise provides an appropriate basis for evaluating the potential effects of changing coastal water levels over the facility's licensing period. Therefore, the NRC staff finds that the applicant's treatment of antecedent water level and sea-level rise is appropriate for the design-basis flood assessment.

The applicant described the strengths and limitations of each modeling approach and explained that SLOSH was used to efficiently evaluate numerous storm scenarios, while ADCIRC/SWAN was used to provide higher-resolution simulation of storm surge propagation, local topography, and wave effects. The applicant stated that the refinement analyses confirmed the screening results and that wave runup effects were subsequently incorporated using the coupled ADCIRC/SWAN modeling approach.

The NRC staff reviewed the applicant's storm surge modeling methodology, model selection, and hierarchical assessment process. The NRC staff finds the applicant's methods and processes acceptable because they are consistent with the hierarchical hazard assessment approach described in NUREG/CR-7046 and the guidance in NUREG-0800, section 2.4.5. The NRC staff finds the use of the SLOSH model for screening acceptable because it provides an efficient means of evaluating a broad range of hurricane scenarios and identifying limiting events. The NRC staff also finds the use of the ADCIRC model coupled with SWAN acceptable because the coupled modeling system provides a refined representation of storm surge

propagation, local hydrodynamics, and wave effects within the San Antonio Bay, Matagorda Bay, and Coloma Creek system. The NRC staff concludes that the applicant's modeling approach provides an acceptable technical basis for determining the design-basis PMSS at the LMGS site.

Generation of the Initial Storm Set

The applicant developed an initial set of synthetic PMH storms for use in the screening-level storm surge analysis. The applicant generated the initial storm set using the PMH parameters described in PSAR section 2.4.5.1, including the radius of maximum winds, forward speed, storm bearing, central pressure deficit, and landfall location. The applicant selected eight landfall locations between NWS Report 23 mile posts 230 and 300 and nine storm bearings spanning 80 to 200 degrees to develop 72 storm tracks representative of storms capable of producing the largest surge at the LMGS site. Each storm track was expanded by varying the forward speed and radius of maximum winds over the applicable PMH parameter ranges, resulting in 1,440 unique synthetic storm scenarios for the screening analysis.

The applicant stated that the selected landfall locations and storm bearings were chosen to maximize the potential for storm surge propagation through the Matagorda Bay, San Antonio Bay, and Coloma Creek system toward the LMGS site while remaining consistent with the PMH parameter ranges established in NWS Report 23.

The NRC staff reviewed the methodology used to develop the initial storm set and finds the applicant's methods and processes acceptable because the applicant systematically evaluated the range of controlling PMH parameters identified in PSAR section 2.4.5.1 and developed a comprehensive set of synthetic storm scenarios representative of the range of credible design-basis hurricanes that could affect the site. The NRC staff also finds that the applicant's selection of landfall locations, storm bearings, forward speeds, and radii of maximum winds provides an adequate technical basis for identifying the combinations of storm characteristics most likely to produce the limiting storm surge at the LMGS site.

The applicant generated an initial storm set using the PMH parameters and ranges described in PSAR section 2.4.5.1. The applicant varied storm bearing, landfall location, radius of maximum winds, forward speed, and central pressure deficit.

The NRC staff reviewed the storm track and parameter selection methodology and determined that the applicant adequately considered the range of storm characteristics capable of producing significant surge at the LMGS site.

Based on the discussion above, the NRC staff concludes that the applicant’s generation of the initial storm set is acceptable for the CP stage.

Adjustment of Central Pressure Deficit

The applicant adjusted the central pressure deficit used in the SLOSH simulations when necessary to ensure that the resulting wind speeds did not exceed the maximum PMH wind speed established for the site region. Because SLOSH internally computes the hurricane wind field using the central pressure deficit and radius of maximum winds, the applicant performed sensitivity analyses to verify that the resulting wind speeds remained within the maximum sustained wind speed established for the PMH. Based on these analyses, the applicant

adjusted the central pressure deficit for selected synthetic storms through an iterative process so that the calculated maximum wind speeds did not exceed the PMH wind speed identified in PSAR section 2.4.5.1. The adjusted central pressure deficits ranged from 100 millibars to the NWS Report 23 value of 133.1 millibars.

The NRC staff reviewed the applicant's methodology for adjusting the central pressure deficit used as input to the SLOSH model. The NRC staff finds the applicant's methods and processes acceptable because the adjustments were made to ensure consistency between the meteorological parameters used by the SLOSH model and the PMH wind characteristics established in accordance with NWS Report 23. The NRC staff finds that limiting the modeled wind field to the established PMH maximum sustained wind speed provides a technically appropriate approach for maintaining consistency among the PMH parameters used in the screening-level storm surge analysis.

Based on the discussion above, the NRC staff concludes that the adjustment of the central pressure deficit is acceptable for the CP stage.

Calculation of the Antecedent Water Level

The applicant calculated the antecedent water level (AWL) in accordance with the guidance in NUREG/CR-7046, ANSI/ANS-2.8-1992, RG 1.59, Revision 2, and ISG, JLD-ISG-2012-06 “Performing a Tsunami, Surge, or Seiche Hazard Assessment” (ML12314A412). The applicant determined the AWL by combining the 10-percent exceedance high tide with projected long-term sea-level rise over the plant operating life. The 10-percent exceedance high tide was calculated using 21 years of observed monthly maximum tide data from the NOAA tide gauge at Rockport, Texas. The applicant then added projected sea-level rise based on the historical rate of sea-level change measured at the same station to obtain a total AWL of 4.0 feet (1.2 m) NAVD 88.

The applicant calculated the AWL using the 10-percent exceedance high tide at the Rockport, Texas, NOAA tide gauge and an assumed 60-year sea-level rise of approximately 1.2 ft (0.37 m).

The resulting AWL used in the PMSS analysis was 4.0 ft (1.2 m) NAVD 88.

The NRC staff reviewed the applicant's treatment of the AWL and sea-level rise and determined that the approach is consistent with the design-basis flood assessment methodology described in the applicant's analysis and applicable NRC guidance.

The NRC staff reviewed the applicant's methodology for determining the AWL and finds the applicant's methods and processes acceptable because they are consistent with NRC guidance for determining the AWL to be combined with the PMSS. The NRC staff finds that the applicant appropriately used a continuous period of observed tide data to determine the 10-percent exceedance high tide and appropriately incorporated projected long-term sea-level rise over the design life of the facility in accordance with applicable NRC guidance. Accordingly, the NRC staff concludes that the applicant established an appropriate AWL for use in the storm surge analyses.

The NRC staff concludes that the calculation of the AWL is acceptable for the CP stage.

SLOSH Model Verification

The applicant verified the SLOSH model using Hurricanes Harvey and Carla. The applicant compared SLOSH-predicted storm surge elevations with observed water levels from Hurricane Harvey (2017) at NOAA tide gauge stations located at Seadrift, Port O'Connor, Port Lavaca, Aransas Wildlife Refuge, and Rockport. The applicant also compared SLOSH results for Hurricane Carla (1961) with documented historical storm surge elevations and high-water marks because continuous tide gauge records were unavailable for that event.

The applicant evaluated differences between modeled and observed peak water levels and the timing of peak surge and concluded that the model generally reproduced the observed regional storm surge characteristics. Based on the Hurricane Harvey comparisons and published evaluations of SLOSH performance, the applicant concluded that the model uncertainty for maximum water surface elevation ranged from approximately -8 percent to +20 percent. To account for this uncertainty during the screening-level assessment, the applicant increased the computed maximum water levels by 10 percent.

The NRC staff reviewed the applicant's verification methodology and comparison of model predictions with historical observations. The NRC staff finds the applicant's methods and processes acceptable because the applicant verified model performance using representative historical hurricanes that affected the site region and compared predicted storm surge elevations with available measured water levels and documented high-water marks. The NRC staff also finds the applicant's incorporation of an uncertainty adjustment into the screening-level analysis acceptable because the adjustment accounts for identified model uncertainty and provides additional conservatism when selecting storms for subsequent refinement-level evaluation.

The NRC staff notes that the applicant did not rely solely on SLOSH for the final PMSS determination but performed refined ADCIRC+SWAN analyses for the selected controlling storm scenarios.

Based on the discussion above, the NRC staff concludes that the SLOSH model verification and its use for screening purposes are acceptable for the CP stage.

Screening-Level Assessment (SLOSH)

The applicant performed a screening-level storm surge assessment using the SLOSH model and the initial storm set to identify the combinations of hurricane parameters that produce the largest storm surge at the LMGS site. The applicant performed 1,440 steady-state storm simulations using the AWL as the initial water surface elevation. Model inputs included the operational NOAA SLOSH basin, AWL, storm direction, landfall location, forward speed, radius of maximum winds, and central pressure deficit.

The applicant evaluated the maximum water surface elevation predicted at the model cell representing the LMGS site for each synthetic storm. The applicant reported that 761 simulations did not produce flooding at the site and that the remaining simulations produced maximum water surface elevations ranging from approximately 23 ft to 37.7 ft (7.01 to 11.5 m) NAVD 88 before application of the model uncertainty adjustment. Based on the screening results, the applicant selected 15 representative storm scenarios that bound the range of significant flooding conditions for further evaluation using the refinement-level model.

The NRC staff reviewed the applicant's screening-level methodology and storm selection process and finds them acceptable because the applicant systematically evaluated a comprehensive range of synthetic PMH scenarios using an established storm surge model and identified the storm parameter combinations that produce the largest surge at the site. The NRC staff also finds the applicant's selection of representative limiting storms for refinement-level analysis acceptable because the selection was based on objective screening criteria and ensured that the subsequent detailed analyses focused on the storms with the greatest potential to produce the design-basis storm surge.

The NRC staff concludes that the screening-level SLOSH assessment is acceptable for the CP stage.

Refined PMSS Analysis

The applicant performed a refinement-level PMSS analysis using the ADCIRC+SWAN coupled hydrodynamic and wave models. The applicant developed an enhanced computational grid based on the NOAA Hurricane Surge On-Demand Forecast System mesh by increasing the spatial resolution within the Matagorda Bay, Powderhorn Lake, and LMGS site vicinity and incorporating updated bathymetric and topographic data from the NOAA Continuously Updated DEM. The applicant also relocated the inland model boundary farther from the site and refined representation of important hydraulic features, including channels, barrier islands, and coastal structures.

The applicant calibrated and verified the ADCIRC+SWAN model using Hurricanes Harvey (2017) and Nicholas (2021). The applicant compared predicted storm surge elevations with observed NOAA tide gauge measurements and compared predicted significant wave heights with observations from NOAA National Data Buoy Center wave buoys. The applicant concluded that inclusion of the SWAN wave model substantially improved agreement between modeled and observed water levels by accounting for wave setup that was not represented in ADCIRC alone.

Following model calibration and verification, the applicant simulated the 15 limiting storm scenarios identified during the screening-level assessment. The applicant incorporated the AWL through a constant sea surface elevation offset representing the combined effects of the 10-percent exceedance tide and projected sea-level rise. The applicant evaluated maximum water surface elevations, significant wave heights, wave directions, and wind speeds throughout the model domain. The applicant identified Storm No. 260 as producing the controlling PMSS at the LMGS site, with a maximum water surface elevation of 36.38 ft (11.09 m) NAVD 88. The applicant also evaluated the duration of flooding above the site grade and concluded that the controlling event would produce floodwaters above the site grade for approximately eight hours.

The NRC staff reviewed the applicant's refinement-level modeling methodology, computational grid development, model calibration, verification, and production simulations. The NRC staff finds the applicant's methods and processes acceptable because the ADCIRC+SWAN modeling system is a technically appropriate state-of-the-art hydrodynamic modeling approach for evaluating coastal storm surge and wave effects. The NRC staff finds that the applicant appropriately refined the computational mesh in areas influencing storm surge propagation to the LMGS site and incorporated current bathymetric and topographic information to improve representation of local hydraulic features. The NRC staff also finds that calibration and verification against Hurricanes Harvey and Nicholas provide reasonable assurance that the

model adequately reproduces observed storm surge and wave conditions in the vicinity of the site.

The NRC staff further finds that the applicant's use of the refined model to evaluate the limiting storm scenarios identified during the screening-level assessment is consistent with the hierarchical hazard assessment approach described in NUREG/CR-7046. The NRC staff concludes that the applicant's refinement-level analyses provide an acceptable technical basis for determining the controlling PMSS, associated wave effects, and flood duration at the LMGS site. Accordingly, the NRC staff finds the applicant's methods and processes acceptable for establishing the design-basis storm surge hazard for the site.

The NRC staff reviewed the refined PMSS analysis and determined that the applicant's methodology provides an adequate basis for establishing the PMSS stillwater elevation at the CP stage.

The staff concludes that the applicant's refined PMSS analysis is acceptable for the CP stage and that the maximum PMSS stillwater elevation of 36.38 ft (11.09 m) NAVD 88 should be used as the design-basis stillwater flood level at the LMGS site.

Seiches and Resonance

The applicant evaluated the potential for flooding at the LMGS site due to seiches and resonance. The applicant stated that a seiche is an oscillation of the water surface within an enclosed or semi-enclosed body of water initiated by an external forcing mechanism. The applicant determined that, with the exception of the plant cooling basin, the LMGS site is not located adjacent to a sufficiently large enclosed or semi-enclosed water body capable of generating seiches that could affect the site. The applicant also stated that although seismic seiches have been observed within portions of the Gulf of America (formerly known as Gulf of Mexico) and associated barrier island systems, the resulting water level oscillations are too small to affect SR or NSRST structures at the site. Based on these considerations, the applicant concluded that seiche-induced flooding is not a credible flooding mechanism for the LMGS site.

The NRC staff reviewed the applicant's evaluation of seiches and resonance and finds the applicant's methods and processes acceptable because the applicant appropriately considered the physical characteristics of the site and surrounding water bodies in evaluating the potential for seiche generation. The NRC staff also finds that the applicant provided an adequate technical basis for concluding that neither atmospheric nor seismic seiches are capable of producing flooding that could challenge the SR or NSRST structures at the LMGS site. Accordingly, the NRC staff concludes that seiche-induced flooding is not a credible design-basis flood mechanism for the LMGS site.

Wave Action

The applicant used the coupled ADCIRC+SWAN model to evaluate wind-generated wave effects associated with the controlling PMSS storm. The applicant calculated a significant wave height of approximately 4.04 ft (1.23 m) and a peak period of approximately 2.8 seconds at the site. The applicant determined the significant wave height and peak wave period corresponding to the controlling storm surge event and used these results to evaluate wave-induced loads on SR structures.

The applicant calculated wave runup and wave forces using the methodology presented in the U.S. Army Corps of Engineers (USACE) Coastal Engineering Manual (CEM). The applicant stated that the methodology assumes waves do not undergo local breaking immediately adjacent to the structures, which the applicant justified based on the shielding effects of surrounding buildings and the wave transformation results produced by the SWAN model. The applicant used the Goda method to calculate wave pressures and wave runup on vertical walls, including the effects of obliquely incident waves, and combined the calculated hydrodynamic wave pressures with hydrostatic pressures to determine the total loading on SR structures.

The NRC staff reviewed the applicant's evaluation of wave action, including the treatment of wave height, wave direction, hydrostatic pressure, and hydrodynamic pressure.

The NRC staff finds that the applicant appropriately considered wind-generated wave effects as a component of the PMSS design-basis flood hazard. This is important because the NRC's flood protection criteria require consideration of natural phenomena and associated effects, rather than consideration of stillwater elevation alone.

The NRC staff concludes that the applicant's evaluation of wave action is acceptable.

Wave Runup

The applicant evaluated wave runup associated with the controlling PMSS using the methodology described in USACE Engineering Manual (EM) 1110-2-1614 and the CEM. The applicant calculated the design wave height, defined as the average height of the highest one percent of waves, from the significant wave height predicted by the SWAN model. The applicant then applied the Goda equations for wave runup on vertical structures, conservatively assuming normal wave incidence to maximize the calculated runup elevation. The applicant defined the peak total water surface elevation at the site as the sum of the stillwater elevation and the calculated wave runup.

The NRC staff reviewed the applicant's methodology for determining wave runup and finds the applicant's methods and processes acceptable because the methodology is consistent with accepted coastal engineering practice and the guidance contained in USACE EM 1110-2-1614 and the CEM. The NRC staff finds that the applicant appropriately used the significant wave conditions developed from the coupled ADCIRC+SWAN simulations as input to the wave runup calculations and conservatively assumed wave conditions that maximize runup on critical structures. Accordingly, the NRC staff concludes that the applicant's methodology provides an acceptable technical basis for determining the maximum total water surface elevation resulting from the combined effects of storm surge and wave runup at the LMGS site.

The applicant calculated wave runup using the USACE methodology and the Goda equations for vertical building surfaces. The applicant combined the calculated wave runup with the PMSS stillwater elevation to establish the maximum total water level at the building walls.

The applicant determined a maximum total water level of 46.49 ft (14.17 m) NAVD 88 at the building walls.

The NRC staff reviewed the methodology used to calculate wave runup and the treatment of the combined stillwater and wave effects. The NRC staff finds that the applicant appropriately considered wave runup as an additional flood effect at building surfaces.

Because wave runup depends on the final geometry, orientation, elevation, surface characteristics, and configuration of the structures and surrounding site, and site grading design and passive engineered flooding protection feature design were unavailable at the time of CP, the NRC staff proposed a site grading and passive engineered flooding protection design permit condition, which requires that LME notify the NRC staff once these aspects of the design and analysis are available.

Stillwater Level and Total Water Level due to Storm Surge

The applicant determined the design-basis PMSS stillwater level at the LMGS site using the hierarchical storm surge assessment described in PSAR sections 2.4.5.2.2.1 through 2.4.5.2.2.9. The applicant stated that the controlling storm produces a maximum PMSS stillwater elevation of 36.38 ft (11.1 m) NAVD 88 at the LMGS site before consideration of wave runup.

The applicant further evaluated wave effects associated with the controlling PMSS event using the coupled ADCIRC+SWAN modeling system and wave runup calculations based on the U.S. Army Corps of Engineers CEM. The applicant stated that the combined effects of storm surge and wave runup are applicable at the walls of the SR and certain NSRST buildings and result in a maximum PMSS flood water level of 46.49 ft (14.17 m) NAVD 88.

The NRC staff reviewed the applicant's determination of the PMSS stillwater level and the maximum water level including wave runup provided in the PSAR and its subsequent revised markups, responses to audit discussion questions and technical reports provided during the regulatory audit.

The NRC staff finds the applicant's methods and processes acceptable because the applicant used a hierarchical modeling approach consisting of screening-level SLOSH simulations followed by refinement-level ADCIRC+SWAN analyses to determine the controlling storm surge, incorporated the AWL in accordance with NRC guidance, and calculated wave runup using accepted coastal engineering methodologies. The NRC staff further finds that the applicant appropriately combined the maximum stillwater elevation with the calculated wave runup to determine the design-basis total water level applicable to SR and NSRST structures. Therefore, the NRC staff concludes that the applicant has established an acceptable technical basis for determining the design-basis stillwater level and total water level due to storm surge at the LMGS site.

As discussed above, the applicant employed a hierarchical hazard assessment consisting of screening-level SLOSH simulations followed by refinement-level ADCIRC+SWAN analyses, incorporated the AWL consistent with NRC guidance, and calculated wave runup using methodologies described in the U.S. Army Corps of Engineers CEM. The NRC staff finds the applicant's methods and processes acceptable for determining the PMSS stillwater level and the maximum water level including wave runup at the CP stage.

The NRC staff agrees that the calculated wave runup is episodic in nature and is associated with transient wave impacts on the exterior walls of structures rather than a sustained site-wide flood level. Therefore, the NRC staff concludes that the PMSS stillwater elevation, together with the associated hydrodynamic effects, represents the appropriate design-basis flood hazard level for the LMGS site.

The NRC staff further concludes that the PMSS stillwater elevation is significantly higher than the finished site grade elevation of 31.5 ft (9.6 m) NAVD 88. Therefore, the applicant should describe the flood protection and flood mitigation features relied upon to protect SR and certain NSRST structures and systems in PSAR section 2.4.10 and describe the associated flood response and emergency procedures in PSAR section 2.4.14, to demonstrate that the design-basis flood can be adequately managed.

In addition, the NRC staff finds that the applicant appropriately considered the proposed engineered backfill materials, their hydraulic properties, and the anticipated duration of inundation in evaluating flood propagation and site flooding behavior. The NRC staff concludes that these considerations provide an acceptable basis for evaluating the site's hydraulic response to the design-basis storm surge.

The NRC staff notes that the final design and construction details of the passive flood protection features may affect local flood propagation and flow patterns at the site. The site grading and passive engineered flooding protection design construction permit condition requires that LME notify the NRC staff once these aspects of the design and analysis are available.

The applicant should revise the PMSS flood hazard analysis at the OL stage to incorporate the final site grading, drainage features, passive flood protection structures, engineered backfill, and any other site configuration changes that could affect storm surge propagation, wave effects, flood elevations, or flood protection of SR and certain NSRST SSCs.

The revised analysis should confirm that the assumptions, methodologies, and boundary conditions used in the PMSS analysis remain applicable to the final configuration. The analysis should include detailed characterization of the PMSS flood hazard, including the effects of storm surge propagation, stillwater elevation, wave action, wave runup, inundation duration, and associated flood effects at the site.

Prior to fuel load, the applicant should verify that the final design and construction of the passive flood protection features, including final site grading, engineered backfill, flood barriers, drainage features, and other permanent flood protection structures credited in the design-basis flood hazard evaluation, are consistent with the assumptions, hydraulic parameters, and boundary conditions used in the PMSS analysis.

The applicant should document the results of this evaluation and provide the information for NRC inspection as part of the OL review.

2.4.5.2 Conclusions

The NRC staff reviewed the applicant's evaluation of probable maximum surge and seiche flooding for the LMGS site, including the PMH meteorological parameters, historical hurricane information, storm surge modeling, AWL, SLOSH screening analysis, refined ADCIRC+SWAN analysis, model calibration and verification, seiche potential, wave action, and wave runup.

The NRC staff determined that the applicant used an appropriate progressive modeling approach to characterize the PMSS hazard at the LMGS site. The NRC staff finds the applicant's use of historical hurricane data and model verification to be reasonable and finds the refined ADCIRC+SWAN analysis adequate to establish the controlling PMSS hazard at the CP stage.

Based on its review, the NRC staff determined that the applicant has provided sufficient information at the CP stage to characterize the PMSS hazard at the LMGS site. The applicant's analysis identifies a maximum PMSS stillwater elevation of 36.38 ft (11.09 m) NAVD 88 at the site. When wave runup is included, the applicant calculates a maximum total water level of 46.49 ft (14.17 m) NAVD 88 at the walls of the site buildings. The NRC staff considers the resulting total water level, including applicable wave effects, to represent the design-basis flood hazard that must be considered in the final design and protection of SR and certain NSRST SSCs.

The NRC staff notes that the final site grading, drainage configuration, and final configuration of passive flood protection features are not available at the CP stage. The site grading and passive engineered flooding protection design construction permit condition requires that LME notify the NRC staff once these aspects of the design and analysis are available and the final analysis and final design information would be required by regulation for an OL application. NRC staff determined that, consistent with the guidance in NEI 21-07, the information provided in the PSAR for PMSS is sufficient to support demonstrating the viability of the design and for issuance of the CP. For additional information on how the PMSS, as the design-basis hazard level for flooding, is addressed in the design (see PSAR section 6.1 and the technical evaluation in section 6.1 of this SE).

2.4.6 Tsunami Hazards

This section describes flooding at the site resulting from tsunami flooding at the site. The NRC staff review assesses the various tsunamigenic sources, along with the probable maximum tsunami evaluations, developed using site-specific geologic, topographic, geomorphic, bathymetric, and hydrologic characteristics incorporated into the model. The NRC staff assessed whether the tsunami hazard represented a design-basis hazard for the site.

The applicant evaluated the potential tsunami hazard for the LMGS site to determine whether a probable maximum tsunami (PMT) could affect the LMGS site. The applicant based the evaluation on published tsunami hazard assessments for the Gulf of America (formerly known as Gulf of Mexico) and identified the maximum credible tsunami-generating mechanism applicable to the Texas Gulf Coast. The applicant evaluated potential tsunamigenic sources, historical tsunami events, conservative source characteristics, and published tsunami modeling results to estimate the maximum tsunami water level that could affect the vicinity of the LMGS site. The applicant combined the estimated tsunami amplitude with the applicable AWL components, including the 10-percent exceedance astronomical high tide, sea-level anomaly, wind setup, and wave runup, to estimate the PMT water level at the site. The applicant also identified conservatisms incorporated into the analysis.

The applicant stated that the LMGS site is located approximately 22 mi (35.4 km) inland from the Gulf Coast and approximately 4 mi (6.4 km) northwest of Mission Lake, with a site elevation that is substantially higher than the estimated PMT water levels. Based on these evaluations, the applicant concluded that tsunami-induced flooding would not affect SR SSCs at the LMGS site.

2.4.6.1 Technical Evaluation

The NRC staff reviewed the applicant's evaluation of tsunami hazards, including the identification of credible tsunami-generating sources, the use of published Gulf of America

tsunami hazard studies, the consideration of historical tsunami events, and the methodology used to estimate the PMT water levels at the site. The NRC staff finds the applicant's methods and processes acceptable because they are consistent with the guidance in NUREG-0800, section 2.4.6, and applicable NRC guidance for evaluating tsunami hazards. The NRC staff finds that the applicant appropriately considered the regional tsunami hazard, applied conservative assumptions in estimating the PMT water levels, and evaluated the effects of AWL conditions. Based on the NRC staff review of the applicant's analysis and the site's inland location and elevation, the NRC staff concludes that tsunami-induced flooding does not represent a credible hazard to the LMGS site. Accordingly, the NRC staff finds the applicant's tsunami hazard evaluation acceptable.

2.4.6.1.1 Probable Maximum Tsunami

The applicant evaluated the potential tsunami hazard for the LMGS site to determine whether a PMT could affect the safety-significant SSCs. The applicant based the evaluation on published tsunami hazard assessments for the Gulf of America and identified the maximum credible tsunami-generating mechanism applicable to the Texas Gulf Coast. The applicant evaluated potential tsunamigenic sources, historical tsunami events, conservative source characteristics, and published tsunami modeling results to estimate the maximum tsunami water level that could affect the vicinity of the LMGS site. The applicant combined the estimated tsunami amplitude with the applicable AWL components, including the 10-percent exceedance astronomical high tide, sea-level anomaly, wind setup, and wave runup, to estimate the PMT water level at the LMGS site. The applicant also identified conservatisms incorporated into the analysis.

The applicant stated that the LMGS site is located approximately 22 mi (35 km) inland from the Gulf Coast and approximately 4 mi (6 km) northwest of Mission Lake, with a site elevation that is substantially higher than the estimated PMT water levels. Based on these evaluations, the applicant concluded that tsunami-induced flooding would not affect the LMGS site.

The applicant reviewed published tsunami modeling studies addressing submarine landslides along the U.S. Atlantic Margin, the Puerto Rico Trench, the Azores-Gibraltar fracture zone, and other Atlantic Ocean sources. Based on these studies, the applicant concluded that tsunami waves generated by far-field sources would be substantially attenuated before reaching the Texas Gulf Coast because of wave dispersion across the continental shelf and through the Florida Straits. The applicant also evaluated earthquake-generated tsunamis originating within the Caribbean Basin and the Gulf of America and concluded that these events would produce relatively small tsunami amplitudes along the Texas Gulf Coast. In addition, the applicant reviewed historical observations of seismic seiches associated with the 1964 Alaska earthquake and volcanic tsunami studies and concluded that these mechanisms would not produce water levels capable of affecting the LMGS site.

The applicant identified submarine landslides within the Gulf of America as the controlling tsunami-generating mechanism for the PMT. Specifically, the applicant evaluated published investigations of the East Breaks submarine slump and other large submarine landslide features within the Gulf Basin. Based on published numerical modeling studies performed for the National Tsunami Hazard Mitigation Program, the applicant concluded that a maximum credible near-field submarine landslide provides the most conservative estimate of tsunami hazards for the Texas Gulf Coast. Although the applicant noted that no historical tsunami has been documented from such a landslide within the Gulf Basin, the applicant concluded that

postulating a maximum credible submarine landslide provides a conservative basis for evaluating the PMT at the LMGS site.

The NRC staff reviewed the applicant's identification of credible tsunami-generating mechanisms and the basis for selecting the controlling PMT source. The NRC staff finds the applicant's methods and processes acceptable because the applicant considered the range of credible near-field and far-field tsunami sources identified in published technical literature and evaluated the relative effects of submarine landslides, earthquakes, volcanic events, and seismic seiches on the Texas Gulf Coast. The NRC staff finds that the applicant appropriately relied on published tsunami hazard assessments and numerical modeling studies developed for the Gulf of America and selected a maximum credible near-field submarine landslide as the controlling PMT source because it produces the most conservative tsunami hazard for the site. Accordingly, the NRC staff concludes that the applicant established an acceptable technical basis for identifying the PMT source mechanism for the LMGS site.

The applicant evaluated potential tsunami-generating mechanisms that could affect the LMGS site to identify the PMT for the Texas Gulf Coast. The applicant reviewed published studies (e.g., Ten Brink et al., 2007; Knight, 2006; Mader, 2001) of tsunami hazards in the Gulf of America and Atlantic Ocean and evaluated both near-field and far-field tsunami sources, including submarine landslides, tectonic earthquakes, seismic seiches, volcanic events, and subaerial slope failures.

Far-Field Sources

The applicant identified major far-field submarine landslide zones along the U.S. Atlantic Margin, the Scotia Margin near eastern Canada, the Storegga landslide zone in the North Atlantic east of Iceland, and the Puerto Rico Trench. Of these, submarine landslides along the U.S. Atlantic Margin, from Georges Bank to the Blake Spur near the Carolina Trough, were identified as the most significant potential contributors to tsunami generation affecting the Gulf Coast.

Numerical model simulations (Ten Brink et al., 2007) indicate that tsunami amplitudes along the Atlantic Coast would be considerably reduced due to energy dissipation over the broad continental shelf. Tsunami propagation into the Gulf of America would be further attenuated as waves traverse the shallow Florida Strait. Similar attenuation effects have been demonstrated in modeling of tsunamis originating from the Puerto Rico Trench (Knight, 2006).

Near-Field Sources

Potential near-field tsunamigenic sources were identified within the Gulf of America, which is divided into three major geologic provinces: Carbonate, Salt, and Canyon/Fan. Submarine landslide evidence exists in all three provinces (Ten Brink et al., 2007). The largest identified failures are located in the Canyon/Fan Province (e.g., Bryant Canyon, Mississippi Fan) and the Salt Province (e.g., East Breaks slump).

The applicant noted that most failures outside of the East Breaks and Canyon/Fan regions are related to salt tectonics and are not expected to produce significant tsunamis. However, a postulated landslide at the East Breaks slump was identified as a credible near-field source for the PMT because of its potential to generate higher wave amplitudes. Trabant et al. (2001) estimated an initial tsunami amplitude of approximately 7.6 m for a hypothesized East Breaks

slump event. Subsequent hydrodynamic modeling under the U.S. National Tsunami Hazard Mitigation Program (NTHMP) (Ten Brink et al., 2009) indicated that slides at the East Breaks and Campeche margin could produce similar tsunami amplitudes along the Texas coast, with a potential maximum runup of approximately 4 m above mean sea level.

Earthquake-Generated Tsunamis

The applicant identified potential tsunamigenic earthquake sources in the Caribbean Basin and the East Atlantic Ocean. Caribbean Basin sources include the North Panama Deformation Belt, the Northern South America Convergent Zone, and the Puerto Rico Trench. The Azores-Gibraltar fracture zone, west of Portugal, represents a potential teletsunami source.

Model simulations (Ten Brink et al., 2007) estimated that tsunami amplitudes at the 250-m isobath along the Gulf Coast would not exceed 0.7 m. Modeling by the West Coast and Alaska Tsunami Warning Center (WCATWC) for severe Caribbean earthquake scenarios (Knight, 2006) predicted maximum tsunami amplitudes less than 0.35 m at the Texas shoreline.

The applicant also reviewed the 1755 Great Lisbon Earthquake, which generated the most significant historical teletsunami affecting the Atlantic Basin. Model results (Mader, 2001) indicate that tsunami amplitudes along the Texas Gulf Coast from this event would have been less than 1.0 m.

Earthquakes recorded within the Gulf of America are rare and generally of low magnitude (≤5.8). The U.S. Geological Survey concluded that such mid-plate earthquakes are unlikely to generate destructive tsunamis (NEIC, 2006).

Other Tsunamigenic and Seiche Sources

Seismic seiches associated with distant earthquakes, such as the 1964 Great Alaska Earthquake, have been recorded along the Texas coast. Observations indicate maximum double amplitudes of 0.26 m at Rockport, Texas (McGarr and Vorhis, 1972). These represent near-maximum seiche magnitudes expected for the region (Ten Brink et al., 2009).

No volcanic or subaerial slope-failure mechanisms capable of producing tsunamis are present in the Gulf of America. Although flank failure of the Cumbre Vieja volcano in the Canary Islands has been modeled as a potential Atlantic tsunami source (Ward and Day, 2001), simulations indicate that such an event would not generate destructive wave heights along the U.S. Gulf or East Coasts (Ten Brink et al., 2007).
Based on the available information, the applicant postulated that a near-field submarine landslide within the Gulf Basin, the East Breaks slump represents the bounding source mechanism for the PMT affecting the Texas Gulf Coast.

The NRC staff reviewed the applicant's identification of credible tsunami-generating mechanisms and the basis for selecting the controlling PMT source. The NRC staff finds the applicant's methods and processes acceptable because the applicant considered the range of credible near-field and far-field tsunami sources identified in published technical literature and evaluated the relative effects of submarine landslides, earthquakes, volcanic events, and seismic seiches on the Texas Gulf Coast. The NRC staff finds that the applicant appropriately relied on published tsunami hazard assessments and numerical modeling studies developed for the Gulf of America and selected a maximum credible near-field submarine landslide as the

controlling PMT source because it produces the most conservative tsunami hazard for the LMGS site. Accordingly, the NRC staff concludes that the applicant established an acceptable technical basis for identifying the PMT source mechanism for the LMGS site.

2.4.6.1.2 Historical Tsunami Record

The applicant reviewed historical tsunami records affecting the Texas Gulf Coast using the NOAA NCEI, 2023 Global Historical Tsunami Database and other published references. The applicant identified four historical tsunami or tsunami-related events that affected the Texas Gulf Coast between 1918 and 2022. These events included probable and doubtful tsunami events associated with Caribbean earthquakes, seismic seiches generated by the 1964 Great Alaska Earthquake, and the 2022 Tonga volcanic eruption. The applicant noted that recorded water level fluctuations associated with these events were relatively small, with documented runup or oscillation amplitudes generally less than approximately 2.1 ft (0.64 m), and that no historical evidence of significant tsunami inundation or paleotsunami deposits has been identified along the Texas Gulf Coast.

The applicant evaluated potential tsunamigenic source mechanisms that could affect the Texas Gulf Coast in the vicinity of the LMGS site. Both near-field and far-field sources were considered, including submarine landslides, earthquakes, and volcanic activity, based on published literature and numerical modeling studies (e.g., Ten Brink et al., 2007; Knight, 2006; Mader, 2001).

Far-Field Sources

The applicant identified major far-field submarine landslide zones along the U.S. Atlantic Margin, the Scotia Margin near eastern Canada, the Storegga landslide zone in the North Atlantic east of Iceland, and the Puerto Rico Trench. Among these, submarine landslides along the U.S. Atlantic Margin, from Georges Bank to the Blake Spur near the Carolina Trough, were identified as the most significant potential contributors to tsunami generation capable of affecting the Gulf Coast.

Numerical model simulations (Ten Brink et al., 2007) indicate that tsunami amplitudes along the U.S. Atlantic Coast would be substantially reduced because of dissipation of wave energy over the broad continental shelf. Propagation into the Gulf of America would be further attenuated as waves traverse the shallow Florida Strait. Similar attenuation effects have been shown in the modeling of tsunamis originating from the Puerto Rico Trench (Knight, 2006).

Near-Field Sources

Potential near-field tsunamigenic sources within the Gulf of America were also considered. The Gulf of America is divided into three major geologic provinces; Carbonate, Salt, and Canyon/Fan, all of which exhibit evidence of submarine landslides (Ten Brink et al., 2007). The largest identified failures are located in the Canyon/Fan Province (e.g., Bryant Canyon, Mississippi Fan) and in the Salt Province (e.g., East Breaks slump).

The applicant noted that most slope failures outside of the East Breaks and Canyon/Fan regions are related to salt tectonics and are not expected to produce significant tsunamis. However, a

postulated landslide at the East Breaks slump was identified as the most credible near-field source for the PMT because of its potential to generate relatively large initial wave amplitudes.

Trabant et al. (2001) estimated an initial tsunami amplitude of approximately 7.6 m for a hypothetical East Breaks slump event. Subsequent hydrodynamic modeling conducted under the U.S. NTHMP (Ten Brink et al., 2009) indicated that slides along the East Breaks and Campeche margins could produce comparable tsunami amplitudes along the Texas coast, with a potential maximum runup of approximately 4 m relative to mean sea level.

Earthquake-Generated Tsunamis

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Model simulations (Ten Brink et al., 2007) estimated that tsunami amplitudes at the 250-m isobath along the Gulf Coast would not exceed 0.7 m. Modeling by the WCATWC for severe Caribbean earthquake scenarios (Knight, 2006) predicted maximum tsunami amplitudes less than 0.35 m at the Texas shoreline.

The applicant also reviewed the 1755 Great Lisbon Earthquake, which generated the most significant historical teletsunami affecting the Atlantic Basin. Model results (Mader, 2001) indicate that tsunami amplitudes along the Texas Gulf Coast from this event would have been less than 1.0 m.

Earthquakes recorded within the Gulf of America are rare and of low magnitude (≤ 5.8). The U.S. Geological Survey (NEIC, 2006) concluded that such mid-plate earthquakes are unlikely to generate destructive tsunamis.

Other Tsunamigenic and Seiche Sources

Seismic seiches associated with distant earthquakes, such as the 1964 Great Alaska Earthquake, have been recorded along the Texas coast. Observations indicate maximum double amplitudes of 0.26 m at Rockport, Texas (McGarr and Vorhis, 1972). These amplitudes are considered near the upper limit of seiche magnitudes expected for the region (Ten Brink et al., 2009).

No volcanic or subaerial slope-failure mechanisms capable of producing tsunamis are present within the Gulf of America. Although flank failure of the Cumbre Vieja volcano in the Canary Islands has been modeled as a potential Atlantic tsunami source (Ward and Day, 2001), simulations indicate that such an event would not generate destructive wave heights along the U.S. Gulf or East Coasts (Ten Brink et al., 2007).

The applicant concluded that the historical record demonstrates that tsunami effects along the Texas Gulf Coast have been minor and that no historical event approached elevations capable of affecting the LMGS site.

The NRC staff reviewed the applicant's evaluation of historical tsunami records and finds the applicant's methods and processes acceptable because the applicant used recognized

historical tsunami databases and supporting published literature to characterize documented tsunami events affecting the Texas Gulf Coast. The NRC staff finds that the historical record supports the applicant's conclusion that observed tsunami and tsunami-related water level oscillations in the region have been small and are substantially below elevations that would challenge the LMGS site. The NRC staff concludes that the historical tsunami record provides an acceptable historical basis supporting the applicant's overall PMT hazard assessment.

2.4.6.1.3 Source Generator Characteristics

The applicant evaluated the characteristics of potential submarine landslide sources capable of generating the PMT affecting the Texas Gulf Coast. The applicant postulated that the tsunami source most likely to produce a PMT at the Texas Gulf Coast would be a submarine landslide within the Gulf of America. No historical or instrumental records exist of tsunamis generated from such sources. Although earthquake-generated tsunamis from the Caribbean Basin have been observed and documented, their amplitudes along the Texas Gulf Coast would be considerably smaller than those expected from a hypothetical near-field submarine landslide within the Gulf Basin.

The largest submarine landslide complex mapped by the U.S. Geological Survey (USGS) within the Gulf Basin occurs in the middle and upper Mississippi Canyon/Fan Province (Ten Brink et al., 2007). The complex has an estimated area of approximately 23,000 km², a maximum thickness of about 100 m, and a total displaced volume of approximately 1,725 km³. Geological evidence indicates that this complex has been inactive since the mid-Holocene.

Within the Salt Province and along the base of the Sigsbee Escarpment, thirty-seven distinct submarine landslides have been identified (Ten Brink et al., 2007). The largest of these failures is located offshore of the Rio Grande River in the northwestern Gulf of America (see PSAR figure 2.4.6-2). This feature, known as the East Breaks slump, is interpreted as the result of failure of a shelf-edge delta that formed during the last sea-level lowstand.

Ten Brink et al. (2007) estimated that the East Breaks slump is approximately 114 km in length, 53 km in width, and covers a total area of about 2,250 km². No detailed slide-parameter estimates were provided in that study. Subsequent characterization by Trabant et al. (2001) describes the East Breaks slump as having a head-scarp approximately 20 km wide, located at the ~150 m isobath. The overall slide complex is approximately 3,200 km² in area, with a total erosional length of about 55 km and a maximum slide thickness of about 70 m. The total displaced volume is estimated to be between 50 and 60 km³.
Based on these characteristics, the applicant concluded that a postulated submarine landslide in the East Breaks region represents the most credible and conservative bounding source for generation of the PMT that could affect the Texas Gulf Coast and, by extension, the LMGS site.

The NRC staff reviewed the applicant's characterization of the controlling submarine landslide source and finds the applicant's methods and processes acceptable because the applicant relied on published geological investigations and tsunami hazard studies to establish conservative source characteristics for the PMT evaluation. The NRC staff finds that the applicant appropriately summarized the available information describing the dimensions and location of the East Breaks slump and other significant Gulf of America submarine landslide complexes and used this information consistently with published tsunami hazard assessments. The NRC staff concludes that the applicant established an acceptable technical basis for the source generator characteristics used to evaluate the PMT hazard at the LMGS site.

The NRC staff reviewed the applicant's characterization of the controlling submarine landslide source and finds the applicant's methods and processes acceptable because the applicant relied on published geological investigations and tsunami hazard studies to establish conservative source characteristics for the PMT evaluation. The NRC staff finds that the applicant appropriately summarized the available information describing the dimensions and location of the East Breaks slump and other significant Gulf of America submarine landslide complexes and used this information consistently with published tsunami hazard assessments. The NRC staff concludes that the applicant established an acceptable technical basis for the source generator characteristics used to evaluate the PMT hazard at the LMGS site.

2.4.6.1.4 Tsunami Analysis

The applicant evaluated the PMT hazard for the LMGS site using published tsunami hazard studies rather than performing site-specific numerical modeling. The applicant stated that this approach is appropriate because the site is located approximately 22 mi (35 km) inland from the Gulf Coast at an elevation that is substantially above expected tsunami water levels. The applicant relied primarily on published numerical modeling studies that evaluated tsunami generation, propagation, and inundation within the Gulf of America and along the Texas Gulf Coast.

The NRC staff reviewed the applicant's tsunami hazard assessment methodology and finds the applicant's methods and processes acceptable because the applicant used published, technically supported tsunami hazard studies applicable to the Gulf of America and the NRC staff agrees with the applicant’s determination that additional site-specific numerical modeling was unnecessary based on the site's inland location, elevation, and expected margin above the estimated PMT water levels. The NRC staff concludes that the applicant established an acceptable technical basis for evaluating the tsunami hazard using available published analyses.

2.4.6.1.5 Tsunami Water Levels

The applicant estimated the maximum PMT water level at the LMGS site using results from published probabilistic tsunami modeling studies. The applicant relied primarily on Horillo et al. (2015), which evaluated multiple submarine landslide scenarios throughout the Gulf of America using coupled three-dimensional and two-dimensional hydrodynamic models that were previously validated under the NTHMP. Based on these studies, the applicant selected a conservative tsunami amplitude of 1.5 m within San Antonio Bay for the controlling PMT. To determine the total PMT water level, the applicant combined the conservative tsunami amplitude with the 10-percent exceedance astronomical high tide and projected long-term sea-level rise, consistent with the AWL assumptions used for the storm surge analysis. The applicant then conservatively assumed that the tsunami propagated inland without attenuation, dispersion, or energy loss after passing the coastal barrier system and VBC. The applicant further evaluated coincident wind setup and wave runup associated with a 2-year design wind event using methodologies from ANSI/ANS-2.8-1992, the U.S. Army Corps of Engineers CEM, EM 1110-2-1614, FEMA guidance, and the ACES. The applicant determined the corresponding wind setup, wave characteristics, wave runup, and total PMT water level and concluded that the resulting total tsunami elevation remains below both the existing site grade and the finished floor elevation of the LMGS structures. The applicant also evaluated tsunami-induced drawdown and

concluded that reduced water levels would not affect SR or NSRST SSCs because the plant does not depend on San Antonio Bay as a water source.

The applicant evaluated the effects of the PMT on SR SSCs using the conservatively estimated total tsunami water level, including AWL, wind setup, and wave runup. The applicant determined that the PMT stillwater level of 2.7 m NAVD 88, when combined with 1.79 ft (0.546 m) wind setup and 6.37 ft (1.94 m) wave runup, resulting in a total water elevation of approximately 17.06 ft (5.200 m) NAVD 88. This level is well below the LMGS site grade of 27 ft (8.2 m), and substantially below the NI grade of approximately 31 ft (9.4 m) NAVD 88.

The NRC staff reviewed the applicant's evaluation of tsunami water levels and associated effects and finds the applicant's methods and processes acceptable because the applicant used published, validated tsunami modeling results together with conservative assumptions for inland wave propagation, AWL, wind setup, and wave runup. The NRC staff finds that the applicant's use of established guidance and engineering methodologies for evaluating coincident wave effects provides a conservative estimate of the total PMT water level at the LMGS site. The NRC staff also finds acceptable the applicant's evaluation of tsunami drawdown because no SR or NSRST SSCs rely on San Antonio Bay as a water source.

Based on its review, the NRC staff concludes that the estimated PMT water level, including AWL, wind setup, and wave runup, remains below the existing site grade and the finished floor elevation of the LMGS structures. Because the PMT combined water level is far below plant grade, the NRC staff concludes that tsunami-induced flooding, debris impact, sediment transport, and associated dynamic effects would not challenge the SR or NSRST SSCs at the LMGS site. Therefore, the NRC staff concludes that the PMT does not constitute a design-basis flood hazard for the LMGS site and would not adversely affect the capability of safety-significant SSCs to perform their intended safety functions.

2.4.6.1.6 Summary of Analysis Conservatisms

The applicant identified several conservative assumptions incorporated into the PMT evaluation. The applicant stated that the controlling tsunami source was selected from published probabilistic studies that account for uncertainties in submarine landslide characteristics and represent maximum credible events for the Gulf of America. The applicant further conservatively assumed that the tsunami propagated inland without attenuation, dispersion, or energy loss after crossing the barrier islands and VBC. In addition, the applicant combined the maximum tsunami amplitude with the 10-percent exceedance astronomical high tide and projected sea-level rise and included coincident wind setup and wave runup using conservative engineering assumptions, including the longest fetch, highest one-percent wave height, no reduction factors, and the most conservative wave angle of incidence.
Based on these assumptions, the applicant determined that the maximum total tsunami elevation remains substantially below the existing site grade and the finished floor elevation of the LMGS structures.

The NRC staff reviewed the applicant's evaluation of conservatisms incorporated into the PMT assessment and finds the applicant's methods and processes acceptable because the applicant used conservative assumptions that intentionally overestimate the potential tsunami hazard. The NRC staff finds that the combined assumptions regarding tsunami source characterization, inland propagation, AWL, wind setup, and wave runup provide reasonable assurance that the

estimated PMT bounds the expected tsunami hazard at the LMGS site. Accordingly, the NRC staff concludes that the applicant's PMT analysis incorporates sufficient conservatism to support the evaluation of tsunami hazards.

2.4.6.1.7 Effects on Safety-Related Facilities

The applicant evaluated the effects of the PMT on SR SSCs using the conservatively estimated total tsunami water level, including AWL, wind setup, and wave runup. Based on this evaluation, the applicant concluded that the maximum tsunami elevation remains well below both the existing site grade and the finished floor elevation of the SR SSCs. Consequently, the applicant concluded that the PMT would not inundate SR SSCs or adversely affect their ability to perform their intended safety functions. The applicant further concluded that because inundation of the site would not occur, tsunami-induced effects such as debris impact, waterborne projectiles, sediment erosion, scour, or deposition would not challenge SR SSCs.

The NRC staff reviewed the applicant's evaluation of the effects of the PMT on SR and certain NSRST facilities and finds the applicant's methods and processes acceptable. The NRC staff's evaluation addressed both SR and certain NSRST SSCs because as described in PSAR section 6.1, certain NSRST structures are credited with protecting SR equipment from flooding hazards. The NRC staff agrees that the estimated total PMT water level remains below the site grade and finished floor elevation of the LMGS structures and, therefore, would not result in flooding of these facilities. The NRC staff further agrees that secondary flood effects, including debris transport, erosion, scour, sediment deposition, and hydrodynamic impacts, are not credible for the PMT because the site would not be inundated. Accordingly, the NRC staff concludes that the PMT is not a design-basis flood hazard for the LMGS site and would not adversely affect the safety functions of SR and NSRST SSCs. Therefore, no additional flood protection or mitigation measures are required for tsunami hazards beyond those already incorporated into the plant design.

2.4.6.2 Conclusions

The NRC staff reviewed the applicant's evaluation of the PMT hazard presented in PSAR section 2.4.6, the supporting technical reports and references, and the applicant's responses to the audit questions regarding the selection of the controlling tsunami source, historical tsunami records, source characterization, published tsunami modeling studies, AWL assumptions, wind setup, wave runup, and evaluation of associated flood effects.

The NRC staff finds the applicant's methods, assumptions, and analytical processes acceptable because they are based on recognized guidance, published tsunami hazard assessments applicable to the Gulf of America, and conservative engineering assumptions that bound the potential tsunami hazard at the LMGS site. The NRC staff agrees with the applicant's use of a maximum credible near-field submarine landslide as the controlling PMT source and finds that the applicant appropriately incorporated conservative assumptions regarding tsunami propagation, AWL, sea-level rise, wind setup, and wave runup.

Based on its review, the NRC staff concurs that the postulated PMT flooding scenario does not result in flooding of the LMGS site. The NRC staff further concludes that the estimated total PMT water level remains below the existing site grade and the finished floor elevation of the LMGS structures. Therefore, the PMT is not a design-basis flood hazard for the LMGS site, and

no additional flood protection or mitigation measures are required for tsunami hazards beyond those incorporated into the plant design.

2.4.7 Ice Effects

2.4.7.1 Technical Evaluation

In this section, the NRC staff reviewed the applicant's evaluation of potential ice effects using historical meteorological and hydrologic records, published references, and site-specific analyses to determine whether ice-related hazards could affect the LMGS site.

2.4.7.1.1 Historical Ice Accumulation

The applicant evaluated long-term meteorological records from nearby weather stations and determined that air temperatures below freezing occur infrequently and are generally limited to short-duration events during the winter months. The applicant also reviewed historical streamflow records, river temperature data, and the U.S. Army Corps of Engineers Ice Jam Database and found no documented occurrences of ice jams or significant river ice formation on the Guadalupe River or San Antonio River.

The applicant analyzed data from two nearby meteorological stations (Victoria and KVCT) which indicate that subfreezing temperatures (below 32°F, 0°C) occur approximately eight days per year, primarily between December and February, with rare occurrences in November and March. Despite these occasional cold events, there is no recorded evidence of ice accumulation on the Guadalupe or San Antonio Rivers, nor any documented ice jams in the region, according to the USACE Ice Jam Database.

Water temperature data from four USGS stations over a 40-year period (1966–2021) show that minimum temperatures in the rivers remain well above freezing, with the coldest recorded water temperatures being 43.7°F (6.5°C). Though minor regional freezing events have been documented (notably in 1940, 1951, 1983, 1989, 2021), these have not resulted in ice jams or significant accumulation affecting river systems near the LMGS site. Thus, the likelihood of recurring ice accumulation in the plant’s vicinity is negligible.

The NRC staff reviewed the applicant's historical evaluation and independently examined available information from the U.S. Army Corps of Engineers Cold Regions Research and Engineering Laboratory (CRREL) Ice Jam Database (NOAA, 2024a and 2024b). The NRC staff confirmed that no documented ice jams have occurred within the vicinity of the LMGS site. The NRC staff finds the applicant's methods and processes acceptable because they are based on long-term meteorological observations, measured water temperature records, and historical ice occurrence databases. The NRC staff concludes that significant historical ice accumulation is not credible at the LMGS site.

2.4.7.1.2 High and Low Water Levels Due to Ice

The NRC staff reviewed the applicant's evaluation of the potential for ice to produce flooding or low water conditions affecting SR and certain NSRST SSCs. The applicant concluded that because river ice and ice jams are not credible at the site, ice-induced flooding and ice-induced reductions in river flow are likewise not credible hazards. The applicant also evaluated the

potential for localized freezing to obstruct small drainage features and noted that the plant flood protection analyses conservatively assume complete blockage of drainage systems during the PMP event.

The NRC staff finds the applicant's methods and processes acceptable because the absence of significant river ice precludes the formation of ice jams capable of producing high or low water conditions. The NRC staff also finds that the applicant's flood analyses conservatively bound any localized drainage blockage caused by freezing conditions. Therefore, the NRC staff concludes that ice-induced high or low water levels do not represent a credible hazard for the LMGS site.

2.4.7.1.3 Ice Sheet Formation

The NRC staff reviewed the applicant’s evaluation of ice sheet formation. The applicant evaluated available information to determine the impacts of ice sheet formation in creating hazards such as flooding or low water conditions at the site. The applicant evaluated the risk of ice sheet formation in water storage basins. The maximum ice thickness was estimated as a function of accumulated freezing degree-days (AFDD) using the modified Stefan equation (2.4.3.7-REF-4 USACE 2004). The AFDD estimates are obtained by summing the freezing degree-days for each day, which is the difference between the freezing point (32°F, 0°C) and the average daily air temperature. The analysis was performed for the water years 1947 through 2023. The estimated maximum AFDD is 39 degree-days occurring on February 2, 1951, with the corresponding ice thickness conservatively estimated to be approximately 5 in (12.7 cm). The NRC staff determined that the estimates presented in PSAR table 2.4.7-1 are consistent with published records of regional freezing events and confirm that ice formation in on-site reservoirs is possible but limited in extent. The small reduction in effective liquid storage volume would not impair system function or plant operation.

2.4.7.1.4 Ice-Induced Forces and Blockages

The NRC staff reviewed the applicant's evaluation of potential ice-induced structural loads and blockages affecting SR SSCs. The applicant concluded that no SR SSCs are subject to ice-induced mechanical loading from river ice, sheet ice, or frazil ice because such conditions are not expected to develop at the LMGS site.

The NRC staff independently reviewed historical records and confirmed that river ice formation and ice jams are not credible in the vicinity of the LMGS site. The NRC staff also considered the regional hydroclimatic conditions and concludes that river ice capable of producing significant structural loads or flow blockages is not expected to occur.

The NRC staff finds the applicant's methods and processes acceptable and concludes that ice-induced structural loading and blockage of safety-significant SSCs are not credible hazards at the LMGS site.

2.4.7.2 Conclusions

The NRC staff reviewed the applicant's ice effects evaluation in the PSAR, supporting technical reports, historical meteorological and hydrologic information, accumulated freezing degree-day analyses, estimated ice thickness calculations, river water temperature records, and responses

provided during the regulatory audit. The NRC staff also independently reviewed available information from the U.S. Army Corps of Engineers CRREL Ice Jam Database to confirm the historical occurrence of river ice and ice jams near the LMGS site.

Based on its review, the NRC staff finds the applicant's methods, assumptions, and analytical processes acceptable, and concludes that:

- Historical records demonstrate no occurrence of ice jams on rivers affecting the LMGS site.
- River water temperatures remain well above freezing and do not support sustained river ice formation.
- Only limited seasonal ice formation is possible within on-site water storage basins, with negligible impact on available storage volume.
- Ice-induced flooding, low water conditions, structural loading, and blockage of safety-significant SSCs are not credible hazards.
- The plant flood protection design conservatively accounts for localized drainage blockage that could occur during rare freezing conditions.

The NRC staff concludes that ice effects are not a credible flooding mechanism at the LMGS site and would not adversely affect the capability of safety-significant SSCs to perform their intended safety functions. Therefore, no additional design features or OL action items are required to address ice-related hazards.

2.4.8 Cooling Water Canals and Reservoirs

2.4.8.1 Technical Evaluation

In this section, the NRC staff reviewed the applicant’s PSAR for the hydraulic design basis developed for canals and reservoirs used to transport and impound water supplied to the safety-significant SSCs.

The applicant stated that the Xe-100 reactor design at LMGS does not rely on cooling water canals or reservoirs to perform RSF. PSAR section 2.4.9.6 describes non-safety-related RWMU drawn from the Guadalupe River through the GBRA Calhoun Canal System. PSAR section 6.4.10 describes the reactor building cooling water system (RBCW), and PSAR section 7.3.9 describes the nuclear island cooling water system (NICW), which provides active, closed-loop cooling to safety-significant and non-safety-significant NI heat loads.

The NRC staff reviewed the evaluation performed by the applicant to estimate the potential of loss of safety-significant water at the LMGS site resulting from loss of function of cooling water canals and reservoirs as well as the design basis for such structures.
The NRC staff reviewed the applicant’s description of the reactor which states, in part, that the Xe-100 reactor design under consideration at the LMGS site does not rely on any cooling water canals and/or reservoirs, including surface water and groundwater sources, to perform any RSFs.

The applicant did not provide additional information supporting the conclusion. The NRC staff reviewed other sections of the application to determine consistency of the information regarding the need for SR cooling water at the LMGS site.

In PSAR section 2.4.9.6 the applicant states that non-safety-related water for the RWMU system is drawn from the Guadalupe River and makeup water is supplied to operating basins from the Guadalupe River via GBRA Calhoun Canal System as described in PSAR section 2.4.1. The LMGS plant condensers are cooled directly by air, thus there will be no need to use water in the cooling process.

In PSAR section 6.4.10 the applicant states that heat generated within the electric motor and conducted from the reactor through the electric motor is removed by the RBCW to establish the motor operational environment.

2.4.8.2 Conclusions

The NRC staff reviewed the information provided by the applicant regarding the Xe-100 design and the NICW in relation to the acceptance criteria of NUREG-0800, SRP section 2.4.8, *Cooling Water Canals and Reservoirs*. The NRC staff's review considered whether cooling water canals, reservoirs, or associated water supply structures are relied upon to support safety-significant SSCs, including systems credited for reactor cooling, decay heat removal, or protection of the reactor core.

The NRC staff finds that the Xe-100 design does not rely on cooling water canals or reservoirs as a source of cooling for safety-significant reactor functions. The reactor core is cooled by a helium primary coolant system, and decay heat removal is accomplished through passive safety features that do not require an external water supply. The NRC staff further finds that the NICW provides cooling for applicable nuclear island auxiliary equipment and does not serve as the primary reactor coolant system, an active emergency core cooling system, or a credited SR ultimate heat sink for decay heat removal.

The NRC staff reviewed the applicant's description of the design features, system functions, and associated cooling requirements and concurred with the applicant's conclusion that cooling water canals and reservoirs are not required to support the credited safety functions of the Xe-100 design. Therefore, the hydraulic, structural, and environmental evaluations associated with cooling water canals and reservoirs described in NUREG-0800, SRP section 2.4.8 are not applicable to the Xe-100 design basis.

Based on the information reviewed, the NRC staff concludes that the applicant has adequately addressed the requirements associated with SRP section 2.4.8 and that the absence of reliance on cooling water canals or reservoirs is acceptable for the Xe-100 design.

2.4.9 Channel Migration or Diversion

2.4.9.1 Technical Evaluation

In this section the NRC staff reviewed the geohydrologic design basis of the plant to ensure that the plant and essential water supplies will not be adversely affected. This review includes stream channel diversions away from the site and stream channel diversions towards the site

(which may lead to flooding). Additionally, in such an event, the applicant needs to show that alternate water supplies are available to safety-significant equipment, if needed.

The NRC staff reviewed the applicant's evaluation of potential channel migration and diversion mechanisms and their potential effects on LMGS. The NRC staff reviewed historical geomorphic information, regional topographic characteristics, climate and hydrologic data, and potential human-induced mechanisms that could result in channel diversion.

The NRC staff's review focused on whether channel diversion could result in either:

- flooding of LMGS facilities; or
- loss of a water supply necessary to perform safety-significant functions.

2.4.9.1.1 Historical Channel Diversions

The NRC staff reviewed the applicant's evaluation of historical channel migration and diversion features near the LMGS site. The applicant evaluated historical information, including U.S. Geological Survey topographic maps and historical river charts dating back to 1846.

The applicant determined that the Guadalupe River channel near the LMGS site has remained generally stable for more than 170 years. Historical geomorphic features, including meander cutoffs and oxbow lakes, indicate gradual channel evolution rather than rapid channel migration or avulsion events.

The NRC staff reviewed the applicant's assessment that the lower Guadalupe River reach is characterized by a low gradient, broad floodplain, natural levees, and decreasing channel sinuosity in the downstream direction. Although localized channel adjustments have occurred within the river valley, the NRC staff confirmed that available historical information does not indicate evidence of abrupt channel diversion or significant avulsion events near the LMGS site.

Based on the review of historical channel behavior, the NRC staff finds that the Guadalupe River near the LMGS site exhibits long-term channel stability and that historical channel migration does not indicate a credible mechanism for adverse effects on the LMGS site.

2.4.9.1.2 Regional Topographic Evidence

The NRC staff reviewed the applicant's description of regional topographic evidence for channel diversion. The applicant states that the morphodynamic characteristics of the rivers in the Gulf Coastal Plain province are strongly influenced by drainage basin size, sediment load, and the long-term sea-level change (Aslan and Blum, 1999 and White and Calnan, 1990).

The LMGS facility does not depend on the Guadalupe River to perform any RSFs. Makeup water is supplied to the non-safety-related RWMU system via an intake located on the GBRA Calhoun Canal, as shown in PSAR figure 2.4.1-10. Flow to the GBRA Calhoun Canal is diverted from the Guadalupe River through a channel situated just upstream of the saltwater barrier near Tivoli, Texas. This diversion point lies downstream of the Vicksburg fault zone, where the river exhibits a gentler gradient and reduced sinuosity, resulting in a more stable and predictable channel compared to upstream reaches.

The Guadalupe River originates in the Edwards Plateau and flows southeast to the Gulf Coastal Plain. Compared to other Texas rivers, its gradient is relatively steep due to regional geologic features, but it carries a lower sediment load. Studies show that sediment accommodation space remains high, indicating that the river is still in the early stages of valley filling and unlikely to undergo sudden avulsions.

The nearest growth fault zone, Sam Fordyce, is over 18 mi (29 km) from the site and does not pose a seismic hazard. Geomorphic responses to this fault include slight changes in river slope and sinuosity upstream, but the downstream of the Guadalupe River downstream near the LMGS site has a more stable profile and channel geometry. No avulsion events linked to fault activity have been recorded near the LMGS site.

The NRC staff reviewed the location of the GBRA diversion point, which is upstream of the saltwater barrier near Tivoli, Texas, and downstream of the Vicksburg fault zone. The applicant identified that this portion of the river exhibits a lower gradient and reduced sinuosity, resulting in a comparatively stable channel configuration.

The applicant also evaluated regional geologic conditions, including sediment accommodation space and nearby growth fault zones. The applicant concluded that the nearest significant fault feature, the Sam Fordyce fault zone, is sufficiently distant from the LMGS site and does not provide a credible mechanism for channel diversion affecting the LMGS site.
The NRC staff reviewed the applicant's evaluation and finds that regional geomorphic conditions do not indicate a credible potential for sudden channel diversion that could adversely affect the LMGS site or required plant functions.

2.4.9.1.3 Ice Causes

A review of historical climate and hydrologic data confirms that there is no record of ice jam events in the Guadalupe River. As also discussed in PSAR section 2.4.2.7, water temperatures remain above freezing, and atmospheric freeze events are infrequent and short-lived.

Given the regional climate and absence of historical evidence, ice-related causes of channel diversion are not considered credible. Therefore, ice-induced flooding or river migration presents no threat to the LMGS site.

The NRC staff reviewed the applicant's evaluation of ice-related causes of channel diversion. Historical climate and hydrologic information indicates that freezing conditions in the region of the LMGS site are infrequent, of short duration, and insufficient to support formation of significant ice jams.

The applicant identified no historical record of ice jam events on the Guadalupe River near the LMGS site. The NRC staff reviewed the regional climate conditions and agrees that ice-induced channel diversion or flooding is not a credible hazard for the LMGS site.

2.4.9.1.4 Flooding of Site Due to Channel Diversion

The NRC staff reviewed the applicant's evaluation of potential flooding resulting from channel diversion toward the LMGS site. The LMGS site lies significantly higher than the floodplain of the Guadalupe River, with a topographic buffer exceeding 16 ft (4.9 m). The applicant states

that modeling of extreme flood events, including the PMF, confirms that floodwaters are contained within the existing river valley.

The applicant also states that for a channel diversion to flood the LMGS site, the river would need to erode and migrate 7 mi (11 km) east. The applicant concludes that this is a geologically improbable event that would occur slowly, allowing time for mitigation. As a result, the applicant concludes, the risk of flooding due to channel diversion remains negligible.

The NRC staff reviewed the applicant's assessment and finds that such a channel migration scenario would require long-term geomorphic processes and is not a credible mechanism for flooding LMGS facilities. The gradual nature of potential channel migration would provide sufficient opportunity for monitoring and implementation of mitigation measures, if required.

2.4.9.1.5 Human-Induced Causes of Channel Diversion

The NRC staff reviewed the applicant's evaluation of human-induced causes that could potentially result in channel diversion. The applicant states that several water control structures exist along the Guadalupe River, with Canyon Dam being the most significant in terms of storage capacity. Located upstream in the Edwards Plateau region, the dam does not mitigate flooding from intense rainfall occurring downstream. Severe floods have occurred in the lower river reach, with the most notable one occurring in October 1998. However, the applicant states, modeling shows that even under PMF conditions or hypothetical dam failure, water levels remain confined within the river valley near the LMGS site. Therefore, the potential for channel diversion leading to flooding of LMGS SR facilities is minimal. The applicant also states that the LMGS site's safety functions are independent of any surface water source, so fluctuating water levels caused by dam operations pose no risk to plant safety.

Other potential human-induced risks to river stability include logjams and sand or gravel mining. Logjams occasionally form near the saltwater barrier at Tivoli, Texas, but are actively managed by the GBRA, reducing their potential to cause channel migration. Although gravel mining has led to river path changes in other Texas river systems, such effects have not been documented along the Guadalupe River. Any impacts from mining would likely be localized and manageable with proper mitigation and therefore do not pose a credible flood risk to LMGS.

The potential for channel diversion caused by emerging energy technologies such as geopressured geothermal development is also considered low. While the Gulf Coast Region (GCR) contains high-pressure, high-temperature geothermal zones, known environmental risks from such projects include induced seismicity and land subsidence. However, these factors are addressed through seismic hazard analysis and do not represent a credible threat of flooding via channel diversion at the LMGS site.

The NRC staff reviewed these evaluations and agreed that these mechanisms do not provide a credible basis for channel diversion affecting the LMGS site.

2.4.9.1.6 Alternate Water Sources

The NRC staff reviewed the applicant's evaluation of alternate water supplies associated with potential channel diversion.

The applicant stated that the Guadalupe River is not required to support any SR function at LMGS. The only plant water supply associated with the Guadalupe River is makeup water for the non-safety-related RWMU system, supplied through the GBRA Calhoun Canal System.

The applicant further stated that the plant condenser cooling system uses direct air cooling, eliminating reliance on a surface water source for heat rejection.

The NRC staff reviewed the applicant's evaluation and finds that potential channel diversion would not result in loss of a safety-significant water supply. Because no safety-significant SSC relies on the Guadalupe River or associated canal system, alternate water supplies are not required to maintain safety-significant functions.

2.4.9.2 Conclusions

The NRC staff reviewed the applicant's evaluation of potential channel migration and diversion mechanisms affecting the LMGS site, including historical channel behavior, regional geomorphic characteristics, ice-related effects, flooding potential, human-induced mechanisms, and the availability of alternate water supplies.

Based on its review, the NRC staff agrees with the applicant's conclusion that channel migration or diversion of the Guadalupe River does not represent a credible hazard to the LMGS site. The NRC staff finds that historical records and regional geomorphic characteristics demonstrate long-term stability of the river channel near the LMGS site and do not indicate conditions conducive to rapid channel diversion or avulsion.

The NRC staff further finds that potential channel migration would occur gradually and would remain within the existing river valley, providing sufficient time for monitoring and implementation of mitigation measures, if necessary. The NRC staff also finds that the LMGS design does not rely on the Guadalupe River or associated canal systems to perform RSF. Therefore, potential diversion of the river channel would not result in loss of a SR water supply.

Based on the information reviewed, the NRC staff concludes that the applicant has adequately followed the guidance in SRP section 2.4.9, *Channel Migration or Diversion*. The NRC staff finds that the potential for channel diversion to adversely affect safety-significant SSCs or safety-significant functions at the LMGS site is acceptably low, and the applicant's evaluation is acceptable.

2.4.10 Flooding Protection Requirements

In this section the NRC staff reviewed the applicant's description of the locations and elevations of LMGS site and those of structures and components required for protection of SR and certain NSRST SSCs compared to design-basis flood conditions to determine if flood effects need to be considered in plant design or in emergency procedures.

2.4.10.1 Technical Evaluation

The NRC staff reviewed the applicant's description of the LMGS flood protection requirements for safety-significant SSCs in PSAR sections 2.4.10 and 6.1 to determine whether they are adequately protected from the design-basis flood hazard. The objective of this review was to

verify that SSCs credited with performing SR and certain NSRST functions remain capable of performing those functions during and following the design-basis flood event.

The NRC staff's review considered the design-basis flood hazard levels (DBHLs), the scope of SSCs requiring flood protection, and the applicant's proposed flood protection strategy. The NRC staff evaluated whether the proposed protection features are consistent with the applicable acceptance criteria, including the guidance provided in RG 1.102, Flood Protection for Nuclear Power Plants.

The NRC staff reviewed the applicant's identification of the controlling external flood hazard, flood elevations, and associated protection criteria. The NRC staff also evaluated the applicant's approach for protecting SSCs credited during and following the design-basis flood event, including consideration of passive and active flood protection features.

The NRC staff also reviewed the applicant's evaluation of flood protection requirements for the LMGS site. The review included the DBHLs established in PSAR section 2.4, the applicability of flood protection requirements to safety-significant SSCs, and the proposed plant-level flood protection strategy.

The NRC staff's review focused on whether the applicant demonstrated that:

1. The controlling DBHL bounds the flood elevations associated with all applicable flood-causing mechanisms.

2. The proposed flood protection strategy provides reasonable assurance that SR and certain NSRST SSCs remain capable of performing their required functions during and following the DBHL flood event.

2.4.10.1.1 Design-Basis Flood Hazard Levels

The NRC staff reviewed the applicant's identification of the DBHLs applicable to the LMGS site. The applicant identified the maximum storm surge as the controlling external flood event for the LMGS site, as described in PSAR section 2.4.5.

The applicant established two DBHLs associated with the maximum storm surge event:

- A maximum stillwater level of elevation 36.38 ft (11.1 m) NAVD 88; and

- A total water level including wave runup of elevation 46.49 ft (14.17 m) NAVD 88.

The applicant stated that the stillwater elevation represents the sustained inundation level used for elevation-based flood protection, while the total water level including wave runup represents transient hydrodynamic loading, splash, and spray effects on building exterior surfaces.

The NRC staff reviewed the applicant's distinction between sustained inundation and transient wave runup effects. The NRC staff finds that the use of the stillwater elevation as the controlling flood elevation for protection against sustained inundation is appropriate because the wave runup elevation represents a localized loading condition affecting exposed building surfaces rather than a sustained flood elevation.

The NRC staff also reviewed the applicant's comparison of flood elevations from the other flood-causing mechanisms evaluated in PSAR sections 2.4.2 through 2.4.9. The NRC staff confirmed that the maximum storm surge DBHL bounds the flood elevations associated with the evaluated mechanisms.

Based on the review, the NRC staff finds that the applicant has adequately established the controlling DBHLs for the LMGS site and that the design-basis flood protection strategy appropriately considers both sustained inundation and transient wave runup effects.

The various flood-causing mechanisms and respective flood depths above the proposed finished grade level are summarized below:

LIP Flooding: Analysis of local PMP indicates a maximum flood depth of 24 in. (0.61 m) above the finished grade resulting in a maximum water surface elevation of 33.5 ft (10.2 m) NAVD 88. Foundations of buildings are 6 in. (0.15 m) above grade, resulting in temporary submergence for approximately 12 hours.

River Flooding (PMF): PMF level at the LMGS site due to PMP in the Lower Guadalupe River are estimated at 38.94 ft (11.87 m) NAVD 88. This flood depth is 5.03 ft (1.53 m) above the riverbank. However, the site itself is not at risk of flooding due to this event, as the overtopped water will disperse into the adjacent low-lying areas which are below the elevation of the LMGS site.

West Coloma Creek PMF: Flooding from West Coloma Creek is estimated to be 33.56 ft (10.23 m) NAVD 88, approximately 2 ft (0.6 m) above the finished floor grade.

Dam-Failure Scenarios: Postulated dam failures in the Guadalupe River Basin is estimated to cause a flood depth of 40.7 ft (12.41 m) NAVD 88. Failure of on-site basin failures (Basin #5 and Basin #31) is estimated to cause flood depth of 32.5 ft (9.91 m), potentially submerging building bases by up to 1 ft (0.31 m).

2.4.10.1.2 Scope of Flood Protection

The NRC staff reviewed the applicant's description of the SSCs subject to design-basis flood protection requirements.

The applicant stated that SR SSCs credited with performing RSFs during or following the design-basis flood event are protected against the controlling DBHLs. The applicant further stated that NSRST and non-safety-related with no special treatment (NST) SSCs are not credited to perform RSF during the design-basis flood event and are therefore not subject to the same protection requirements as SR SSCs. However, as described in PSAR section 6.4.1 and evaluated in section 6.4.1 of this SE, there are certain NSRST structures credited with protecting SR SSCs from the DBHL flooding event.

The applicant stated that SSCs not classified as SR that are credited to support a safety function, or that could adversely affect SR SSC performance during or following the flood event, will be designed to prevent adverse impacts on SR SSCs.

The NRC staff also reviewed the applicant's statement that local flood effects not represented by the controlling external water surface elevation, such as LIP, are evaluated separately through applicable site and SSC design provisions. The NRC staff finds this approach acceptable because these hazards are addressed through the applicable design analyses rather than through the site-wide external flood elevation.

2.4.10.1.3 Flood Protection Strategy

The NRC staff reviewed the applicant's proposed flood protection strategy for LMGS. The applicant stated that the flood protection strategy follows a layered approach consistent with RG 1.102 and relies primarily on passive, elevation-based protection. The NRC staff evaluated the following aspects of the applicant's flood protection strategy.

Elevation-Based and Inundation-Resistant Passive Protection

The applicant identified elevation-based siting as the principal passive flood protection feature for protection against sustained inundation. The applicant stated that SR SSCs will be located above the stillwater DBHL where practicable. Where SSCs cannot be located above the stillwater elevation, incorporated flood protection features will be provided.

The applicant identified potential passive flood protection features, including:

- flood-resistant building construction;
- waterstops at exterior construction joints;
- flood-resistant seals at penetrations; and
- other incorporated building envelope features.

The NRC staff reviewed the applicant's approach and finds that reliance on passive protection features is consistent with the LMGS passive safety design philosophy and provides reasonable assurance that SR SSCs will remain protected during sustained inundation conditions. In addition, the design for NSRST structures to ensure they provide adequate flood protection for the SR inside is described in PSAR section 6.4.1 and evaluated in section 6.4.1 of this SE.

The NRC staff finds that the applicant's approach to providing incorporated flood protection features for SSCs located below the stillwater DBHL is acceptable.

Protection of Building Exterior Surfaces from Wave Runup

The applicant identified wave runup protection as a separate design consideration applicable to building exterior surfaces exposed to the total water level of 46.49 ft (14.17 m) NAVD 88.

The applicant stated that wave runup represents a transient hydrodynamic loading condition, including splash and spray effects, and does not represent a sustained inundation elevation. The applicant identified potential passive building features, including:

- substantially impermeable exterior wall construction;
- water-resistant exterior surfaces;
- flood-resistant penetration seals; and
- flood-resistant exterior opening details.

The NRC staff reviewed the applicant's treatment of wave runup effects and finds that addressing these effects through structural design of building exterior surfaces is appropriate because wave runup represents transient loading rather than a sustained flooding condition.

The NRC staff finds that the applicant's proposed passive protection features are consistent with incorporated flood protection barriers described in RG 1.102 and provide viable path for addressing the DBHL wave runup for affected building structures. Passive engineered flood protection barrier may also be a viable option if NSRST structures that protect SR are not able to withstand the DBHL wave run up.

Active Flood Protection Considerations

The applicant stated that active flood protection measures requiring operator action or mechanical actuation are not credited for protection of SR SSCs from the design-basis flood event.

The applicant further stated that administrative controls and emergency operating procedures may be developed to supplement the passive flood protection strategy but will not be relied upon as the primary means of protecting safety-significant SSCs requiring flood protection.

The NRC staff reviewed the applicant's approach and finds that limiting reliance on active flood protection measures is consistent with the objective of providing reliable protection against external flooding. The NRC staff finds that administrative controls, if implemented, appropriately serve as supplemental measures and are not credited as the primary means of maintaining required SSC operability during the design-basis flood event.

2.4.10.2 Conclusions

The NRC staff reviewed the applicant's evaluation of the LMGS flood protection requirements, including the DBHLs, the scope of SSCs requiring protection, and the proposed flood protection strategy.

The NRC staff finds that the applicant has appropriately identified the maximum storm surge as the controlling design-basis flood event and has established design-basis hazard levels that bound the external flooding hazards evaluated for the site. The NRC staff further finds that the applicant has appropriately distinguished between the sustained inundation level used for elevation-based flood protection and the transient wave runup condition used for structural loading and building exterior surface protection.

The NRC staff reviewed the applicant's flood protection strategy and finds that the proposed approach, which relies primarily on passive elevation-based protection and incorporated flood-resistant building features, is consistent with the guidance in RG 1.102 and with the LMGS passive safety design approach. The NRC staff finds that SSCs credited with performing safety functions during or following the design-basis flood event will be protected, commensurate with their credited functions. SSCs not classified as SR but relied upon to support or protect SR functions, will be protected against the design-basis flood hazard, commensurate with their credited functions.

Based on the evaluation described above, the NRC staff finds that the applicant adequately followed the guidance in NUREG-080, SRP section 2.4.10, Flooding Protection Requirements.

The NRC staff finds that the LMGS flood protection strategy provides a viable design approach for addressing the DBHL flooding event, consistent with NEI 21-07 requirements for the CP stage.

The NRC staff's review was based on the preliminary flood protection design considerations provided by the applicant. Preliminary design for passive flood protection features that may impact site grading were not provided in the PSAR. Based on the flood protection design considerations provided and the site grading and passive engineered flooding protection design permit condition, the NRC staff finds that further technical or design information related to flood protection requirements can reasonably be left for later consideration and will be supplied in the FSAR, consistent with 10 CFR 50.35(a)(2).

The final design should identify the SSCs credited for flood protection or flood response functions and should demonstrate that the incorporated flood protection features, including elevation-based protection, flood-resistant building features, exterior surface protection, penetration seals, and other passive flood barriers, maintain the required flood protection capability for the design-basis flood event.

In addition, the final design should confirm the following:

- Building structures and exterior surfaces subject to wave runup effects are designed to withstand the transient hydrodynamic loading, splash, and spray effects associated with the total water level of 46.49 ft (14.17 m) NAVD 88;

- Any administrative controls or operating procedures developed to supplement the passive flood protection strategy are identified and are not credited as the sole means of maintaining protection of SR SSCs.

2.4.11 Low Water Considerations

In this section, the NRC staff evaluated the applicant's assessment of natural phenomena that could reduce or limit the availability of water supplies required to support SR cooling functions. The objective of the review is to determine whether sufficient water supplies are available to support RSFs, including safe shutdown of the plant, under conditions requiring SR cooling.

2.4.11.1 Technical Evaluation

The NRC staff reviewed the applicant's evaluation of low water considerations for the LMGS site, including the potential effects of drought, low surface water elevations, intake blockage mechanisms, water use limitations, and other natural external hazards that could affect the availability of cooling water supplies.

The applicant stated that LMGS does not rely on cooling water canals or reservoirs, including surface water or groundwater sources, to perform RSFs. The applicant further stated that low water conditions would not adversely affect SR cooling functions.

The NRC staff reviewed the applicant's description of the Xe-100 cooling systems, including the NICW, and evaluated whether safety-significant SSCs rely on external water sources that could be affected by low water conditions. As discussed in PSAR section 2.4.8, the NRC staff finds

that the Xe-100 design does not rely on cooling water canals, reservoirs, or other external water supplies to perform SR reactor cooling functions or provide the ultimate heat sink for decay heat removal.

The NRC staff reviewed the design functions of the Xe-100 reactor cooling systems and finds that reactor core cooling is provided by the helium primary coolant system and that decay heat removal is accomplished through passive safety features independent of external cooling water sources. Therefore, reductions in external water availability resulting from drought, low river levels, or other low water phenomena do not challenge the capability of the reactor to perform its RSF.

The NRC staff also reviewed the applicant's description of the NICW. The applicant states that the NICW provides cooling for nuclear island auxiliary equipment and does not function as the reactor coolant system or as the ultimate heat sink for reactor decay heat removal. The applicant also states that the NICW is an internal plant cooling system and does not require cooling water canals, reservoirs, or raw water supplies for the performance of the credited safety functions associated with reactor cooling.

The NRC staff evaluated the potential for low water conditions to affect auxiliary cooling functions, including potential effects on associated heat exchangers, pumps, or supporting systems. Based on the design approach of the Xe-100, the NRC staff finds that safety-significant functions are not dependent on continuous availability of external cooling water sources. Accordingly, natural events that could result in low water conditions, including drought or reduced surface water elevations, would not result in loss of required cooling capability for safety-significant SSCs.

The NRC staff also reviewed potential low water effects associated with other phenomena evaluated in PSAR sections 2.4.4 through 2.4.9, including dam-related effects, surge or seiche conditions, tsunami effects, icing, and channel migration or diversion. Based on the evaluations performed in those sections, the NRC staff finds that these phenomena do not result in a credible loss of water supply required for SR cooling functions.

The NRC staff reviewed the applicant's discussion of water use limitations and determined that because the plant does not rely on external water sources to perform RSF, Federal, State, or local restrictions on water availability would not adversely affect the capability of safety-significant SSCs to perform their functions.

Based on the information reviewed, the NRC staff finds that low water conditions are not a credible mechanism for preventing safety-significant SSCs from performing their safety functions at LMGS.

2.4.11.2 Conclusions

The NRC staff reviewed the applicant's evaluation of low water considerations for the LMGS site, including potential effects from drought, reduced surface water elevations, intake blockage mechanisms, water use limitations, and other natural external hazards.

The NRC staff finds that the Xe-100 design does not rely on cooling water canals, reservoirs, or external water supplies to perform safety-significant functions associated with reactor cooling or decay heat removal. The NRC staff further finds that the reactor core cooling and decay heat

removal functions are provided through design features that are independent of external cooling water availability.

The NRC staff reviewed the applicant's description of the NICW and concurs that this system provides cooling to applicable nuclear island auxiliary equipment and does not serve as the primary reactor cooling system or the ultimate heat sink for decay heat removal. The NRC staff further finds that the NICW does not require cooling water canals, reservoirs, or raw water supplies that could be adversely affected by low water conditions to perform its credited functions.

The NRC staff also reviewed the potential effects of drought, hydrometeorological events, water control structures, channel migration, icing, and other natural phenomena evaluated in PSAR sections 2.4.4 through 2.4.9. The NRC staff finds that these events do not present a credible mechanism for loss of cooling capability required to maintain plant safety functions.

Based on the information reviewed, the NRC staff concludes that the applicant has adequately followed the guidance in NUREG-0800, SRP section 2.4.11, Low Water Considerations. The NRC staff finds that low water conditions do not adversely affect the capability of safety-significant SSCs to perform their safety functions and that the applicant's evaluation is acceptable.

2.4.12 Groundwater

2.4.12.1 Technical Evaluation

In this section the staff reviewed the applicant's description of hydrogeological characteristics of the LMGS site to ensure that effects of groundwater on plant foundations, reliability of safety-significant water supply and dewatering systems are evaluated using site-specific data and technical analysis.

The NRC staff evaluated the groundwater information provided by the applicant for the LMGS site based on the descriptions, analyses, and supporting evaluations presented in the PSAR. The NRC staff's review also considered supplemental information, calculations, and study reports obtained during audit discussions. The NRC staff evaluated the applicant's characterization of groundwater resources, site-specific hydrogeologic conditions, groundwater flow characteristics, potential subsurface pathways, groundwater modeling approach, and proposed monitoring and safeguards program.

The NRC staff reviewed the applicant's evaluation of groundwater conditions to determine whether the information provided is sufficient to characterize the groundwater system beneath and surrounding the LMGS site and whether the applicant has adequately evaluated potential groundwater pathways associated with postulated releases. The evaluation considered regional hydrogeology, groundwater use, aquifer characteristics, groundwater flow directions and gradients, temporal groundwater variations, hydraulic properties, hydrogeochemical characteristics, and groundwater transport considerations.

The NRC staff finds that the applicant has provided an acceptable description of the groundwater system supporting the LMGS site evaluation. The applicant's evaluation identifies the Gulf Coast Aquifer system as the principal regional aquifer and appropriately characterizes the site-specific water-bearing units beneath the LMGS site as the A, C, and E Sand units of the

Chicot Aquifer. The applicant has demonstrated that these units are separated by low-permeability clay and silt units and that the available hydraulic testing data indicate limited hydraulic communication between the water-bearing zones.

The NRC staff's evaluation of groundwater conditions and potential subsurface pathways is presented in the following subsections.

2.4.12.1.1 Description of On-site Use

The NRC staff reviewed the applicant's description of groundwater use at and surrounding the LMGS site. The evaluation considered regional groundwater availability, existing groundwater users, site-specific groundwater conditions, and the current and projected use of groundwater resources in the vicinity of the proposed facility.

The NRC staff reviewed information provided in the PSAR, supplemented by discussions with the applicant's subject matter experts, and supported documentation available during the regulatory audit. The review focused on whether groundwater resources are used at the LMGS site, whether groundwater withdrawal activities could influence groundwater flow conditions relevant to the safety analysis, and whether potential groundwater users could represent receptors for subsurface transport pathways.

The NRC staff finds that groundwater is not used at the LMGS site for plant operations, process requirements, or potable water supply. The proposed facility relies on alternative water sources for operational needs. Groundwater use in the surrounding area is limited primarily to domestic, irrigation, municipal, and other localized uses. Based on the information reviewed, groundwater withdrawal activities in the vicinity of the LMGS site are not expected to significantly affect site groundwater flow conditions or the evaluation of potential groundwater transport pathways.

Physiography and Geomorphology

The NRC staff reviewed the applicant's characterization of the physiography and geomorphic setting of the LMGS site. The site is located within the Gulf Coastal Plain physiographic province (BEG, 1996), which is characterized by relatively low relief, gently sloping terrain, and sedimentary deposits that thicken and dip toward the Gulf of America (formerly known as Gulf of Mexico).

The NRC staff reviewed information describing the regional geomorphic setting, including surface drainage features, coastal plain deposits, and the relationship between topography and groundwater flow. The applicant identified West Coloma Creek as an important local hydrologic feature that may influence shallow groundwater flow conditions.

Based on the information provided, the NRC staff finds that the applicant has adequately characterized the physiographic and geomorphic conditions relevant to groundwater flow at the LMGS site. The identified topographic gradients and surface drainage features provide an appropriate basis for evaluating groundwater recharge, discharge, and potential groundwater transport pathways.

Regional Groundwater Aquifers

The NRC staff reviewed the applicant's description of regional groundwater resources, including the Gulf Coast Aquifer system and the Chicot Aquifer. The Gulf Coast Aquifer system consists of multiple water-bearing units composed of alternating sands, silts, and clays that extend along the Texas Gulf Coast and dip toward the Gulf of America.

The applicant identified the Chicot Aquifer as the principal regional aquifer relevant to the LMGS site. The NRC staff reviewed the applicant's evaluation of regional groundwater occurrence, recharge mechanisms, and groundwater flow directions. Regional groundwater flow within the Chicot Aquifer is generally toward the southeast and Gulf of America, consistent with regional hydraulic gradients established by topography and aquifer geometry.

The NRC staff finds that the applicant has adequately described the regional aquifer system and appropriately identified the Chicot Aquifer as the relevant groundwater resource for evaluation of site-specific groundwater conditions.

Local Hydrology

The NRC staff reviewed the applicant's evaluation of local hydrologic conditions, including surface water features, groundwater recharge and discharge areas, and interactions between surface water and groundwater.

The applicant identified West Coloma Creek and nearby surface water impoundments as features that may influence local groundwater conditions. The NRC staff evaluated the applicant's assessment that West Coloma Creek may act as a localized groundwater discharge feature for the A Sand under certain conditions. The NRC staff also considered the applicant's evaluation of potential recharge influences from Cooling Water Basins 5 and 6.

The NRC staff finds that the applicant has adequately characterized the local hydrologic setting and appropriately considered interactions between surface water features and shallow groundwater flow conditions. The applicant's evaluation provides an acceptable basis for assessing groundwater movement and potential transport pathways.

Site-Specific Hydrogeology

The NRC staff reviewed the applicant's site-specific hydrogeologic investigation, including subsurface characterization, groundwater monitoring well installation, hydraulic testing, groundwater elevation measurements, and hydrostratigraphic interpretation.

The applicant identified three principal water-bearing units beneath the LMGS site: the A Sand, C Sand, and E Sand units. The NRC staff reviewed the applicant's interpretation that these units are hydraulically separated by intervening low-permeability clay and silt layers. The NRC staff also reviewed the results of slug testing (Brown, et. al., 1995) and aquifer pumping tests performed to evaluate hydraulic properties and connectivity.

The NRC staff finds that the applicant has adequately characterized the site-specific hydrogeologic framework. The available data supports the applicant's conclusion that the A, C, and E Sand units represent distinct groundwater flow zones with limited hydraulic interaction.

Groundwater Sources and Sinks

The NRC staff reviewed the applicant's identification of groundwater sources and sinks affecting the LMGS site. The applicant identified regional recharge areas associated with the outcrop areas of the Gulf Coast Aquifer system and localized groundwater influences associated with surface water bodies and facility-related features.

The NRC staff evaluated the applicant's discussion of groundwater sources and sinks, including potential recharge from Cooling Water Basin No. 5 and localized discharge to West Coloma Creek. The NRC staff finds that the applicant appropriately considered relevant groundwater boundaries and hydrologic features in developing the conceptual groundwater model.

The NRC staff concludes that the identified groundwater sources and sinks are appropriately incorporated into the evaluation of groundwater flow conditions and subsurface transport pathways.

On-site Use of Groundwater

The NRC staff reviewed the applicant's description of on-site groundwater use. The applicant stated that groundwater is not used for LMGS operations, process applications, or potable water supply.

The NRC staff evaluated whether groundwater extraction activities at the LMGS site could influence groundwater gradients or transport analyses. Based on the information reviewed, no operational groundwater withdrawals are planned that would significantly alter groundwater flow conditions beneath the LMGS site.

The NRC staff finds that the applicant has adequately described on-site groundwater use and has demonstrated that groundwater withdrawal activities will not adversely affect the groundwater evaluations supporting the LMGS safety analysis.

2.4.12.1.2 Groundwater Sources

The NRC staff reviewed the applicant's evaluation of groundwater sources in the vicinity of the LMGS site. The review considered historical, current, and projected groundwater use; regional and site-specific groundwater flow characteristics; temporal groundwater trends; aquifer hydraulic properties; and hydrogeochemical conditions. The NRC staff's evaluation was based on information provided in the PSAR, discussions conducted during regulatory audit activities with the applicant's subject matter experts, and supplemental technical information and study reports provided through the electronic reading room.

The NRC staff evaluated whether the applicant adequately characterized groundwater resources potentially relevant to the LMGS site and whether groundwater use or groundwater flow conditions could influence the evaluation of potential subsurface transport pathways. The NRC staff finds that the applicant has provided sufficient information to characterize groundwater sources and groundwater conditions relevant to the LMGS safety analysis.

Historical and Projected Groundwater Use

The NRC staff reviewed the applicant's evaluation of historical, current, and projected groundwater use within the regional area surrounding the LMGS site. The applicant used information from the U.S. EPA, the TWDB, 2024a and 2024b), the Calhoun County Groundwater Conservation District (CCGCD), and other regional groundwater planning documents.

The NRC staff reviewed the applicant's discussion of historical groundwater withdrawals from the Gulf Coast Aquifer system. Groundwater use increased substantially during the mid-twentieth century due primarily to municipal supply and agricultural irrigation demands. In response to groundwater level declines, land subsidence, saltwater intrusion concerns, and impacts to groundwater-dependent resources, groundwater management practices were implemented in portions of the Gulf Coast Aquifer system.

The NRC staff reviewed the applicant's discussion of projected groundwater demand and availability. The applicant identified the South Central Texas Regional Water Planning Area (Region L), Groundwater Management Area 15, and the CCGCD as the applicable regional groundwater planning authorities. The applicant described the use of groundwater availability models developed through the TWDB groundwater management process to evaluate future groundwater availability and desired future conditions.

The NRC staff finds that the applicant appropriately considered historical and projected groundwater use trends and incorporated applicable regional groundwater planning information. The NRC staff concludes that projected groundwater use in the vicinity of the LMGS site is not expected to result in groundwater conditions that would adversely affect the groundwater evaluations supporting the LMGS safety analysis.

Historical Groundwater Use

The NRC staff reviewed the applicant's evaluation of historical groundwater withdrawals from the Gulf Coast Aquifer system. The applicant described groundwater use trends beginning in the early twentieth century and increasing significantly between approximately 1940 and 1960 due to agricultural, industrial, and municipal demands.

The NRC staff reviewed information indicating that historical groundwater withdrawals were concentrated primarily in areas of higher population density and agricultural activity, including portions of the Houston metropolitan area and agricultural regions dependent on groundwater irrigation. The applicant identified groundwater management actions implemented in response to concerns associated with excessive groundwater withdrawal.

The NRC staff finds that the applicant adequately described historical groundwater use patterns and appropriately identified the regional factors affecting groundwater withdrawals. The NRC staff concludes that historical groundwater use information provides an appropriate basis for understanding regional groundwater conditions relevant to the LMGS site.

Current Groundwater Use

The NRC staff reviewed the applicant's evaluation of current groundwater use within the vicinity of the LMGS site. The applicant evaluated groundwater use information from EPA drinking

water databases, TWDB water use records, CCGCD information, and TCEQ public water system records.

The NRC staff reviewed the applicant's identification of groundwater users within approximately 50 mi (80 km) of the LMGS site and within approximately 5 mi (8 km) of the proposed facility. The applicant identified groundwater uses including municipal supply, manufacturing, irrigation, livestock, and domestic supply.

The NRC staff reviewed the applicant's identification of the nearest public water system and nearby groundwater wells. The applicant indicated that the nearest public water system is located several miles northeast of the LMGS site and is not hydraulically positioned to represent a potential receptor for groundwater transport from the LMGS site. The nearest identified domestic or irrigation well is located approximately 0.8 mi (1.2 km) west of the proposed LMGS site.

The NRC staff finds that the applicant has adequately characterized current groundwater use in the vicinity of the LMGS site. Based on the groundwater flow direction, hydraulic gradients, and groundwater velocity evaluations, the NRC staff concludes that existing groundwater users are appropriately considered in the assessment of groundwater pathways and are not expected to be adversely affected by facility operations.

Projected Groundwater Use

The NRC staff reviewed the applicant's evaluation of projected groundwater use and availability. The applicant identified the South Central Texas Regional Water Planning Area (Region L) as the applicable water planning region and described the groundwater availability evaluations performed using TWDB groundwater availability models.

The NRC staff reviewed the applicant's discussion of modeled groundwater availability projections for the Gulf Coast Aquifer. The applicant stated that groundwater supplies available from the Gulf Coast Aquifer in the region are projected to generally increase during the planning period through 2070 and that projected groundwater availability in Calhoun County remains sufficient throughout the planning horizon.

The NRC staff finds that the applicant appropriately incorporated regional groundwater planning information and groundwater availability projections into the groundwater evaluation. The NRC staff concludes that projected groundwater demand and availability will not adversely affect the groundwater flow conditions used in the LMGS groundwater transport assessment.

Groundwater Flow

The NRC staff reviewed the applicant's evaluation of groundwater flow conditions using regional groundwater information, site-specific groundwater monitoring data, potentiometric surface maps, hydraulic gradient calculations, and aquifer testing results.

The NRC staff reviewed information indicating that regional groundwater flow within the Chicot Aquifer is generally toward the southeast and Gulf of America. The applicant identified that local groundwater flow conditions beneath the LMGS site are controlled by the site-specific hydrostratigraphic framework, groundwater elevations, hydraulic gradients, surface water features, and localized facility-related influences.

The NRC staff reviewed groundwater elevation data collected from the LMGS monitoring network installed between 2023 and 2024, supplemented by selected monitoring wells at the adjacent SDO facility. The applicant evaluated groundwater flow directions within the A, C, and E Sand units using potentiometric surface maps prepared from February 2024 through September 2025.

The NRC staff finds that the applicant adequately characterized groundwater flow directions within the site-specific water-bearing units.

A Sand

The NRC staff reviewed the applicant's evaluation indicating that groundwater flow within the A Sand is generally toward the northeast and east, with localized flow convergence toward West Coloma Creek. The applicant identified Cooling Water Basin No. 5 as a potential local influence on A Sand groundwater elevations and flow directions.

The NRC staff reviewed the applicant's hydraulic gradient calculations and groundwater velocity estimates for the A Sand. The NRC staff finds that the applicant appropriately considered the relatively low hydraulic gradients and low groundwater velocities associated with the A Sand.

C Sand

The NRC staff reviewed the applicant's evaluation indicating that groundwater flow within the C Sand is generally from east to west across the LMGS site. The NRC staff reviewed potentiometric surface maps and hydraulic gradient calculations supporting this interpretation.

The NRC staff finds that the applicant adequately characterized C Sand groundwater flow conditions and appropriately considered the relatively stable groundwater elevations observed during the monitoring period.

E Sand

The NRC staff reviewed the applicant's evaluation indicating that groundwater flow within the E Sand is generally toward the south and southeast. The NRC staff considered the applicant's evaluation of groundwater elevations, hydraulic gradients, and aquifer testing results.

The NRC staff finds that the applicant adequately characterized E Sand groundwater flow conditions and appropriately evaluated the limited hydraulic connection between the E Sand and shallower water-bearing zones.

<u>Hydraulic Gradients and Groundwater Velocity</u>

The NRC staff reviewed the applicant's calculation of horizontal and vertical hydraulic gradients and groundwater velocities using measured groundwater elevations and hydraulic conductivity values derived from aquifer testing.

The NRC staff finds that the applicant appropriately evaluated groundwater flow velocities using site-specific hydraulic properties and groundwater gradients. The calculated groundwater velocities support the applicant's conclusion that groundwater transport beneath the LMGS site is relatively slow.

The NRC staff concludes that the applicant has adequately characterized groundwater flow conditions relevant to evaluation of potential subsurface pathways.

Temporal Groundwater Trends

The NRC staff reviewed the applicant's evaluation of temporal groundwater trends using available regional groundwater monitoring data, site-specific groundwater elevation measurements, hydrographs, precipitation data, and historical groundwater monitoring information from the adjacent SDO facility. The NRC staff's review considered seasonal variability, long-term groundwater trends, potential influences from precipitation events, and the stability of groundwater flow conditions over the period of observation.

The NRC staff reviewed regional groundwater elevation information from the TWDB monitoring network and historical groundwater monitoring data from the SDO facility. The applicant identified that regional groundwater elevations have remained relatively stable in the vicinity of the LMGS site, with variations attributable to regional groundwater use patterns, precipitation variability, and localized hydrogeologic conditions.

The NRC staff reviewed site-specific groundwater monitoring data collected from February 2024 through September 2025. The applicant evaluated groundwater elevation trends for the A, C, and E Sand units using manual groundwater level measurements and continuous transducer data.

The NRC staff finds that the applicant has adequately characterized temporal groundwater trends at the LMGS site.

A Sand Temporal Trends

The NRC staff reviewed the applicant's evaluation indicating that groundwater elevations in the A Sand exhibit greater variability than groundwater elevations observed in the C and E Sand units. The applicant attributed observed fluctuations to precipitation patterns, localized recharge influences, and interactions with surface water features.

The NRC staff reviewed the applicant's comparison of A Sand groundwater elevations with precipitation data from KVCT. The evaluation indicates that periods of increased precipitation generally correspond with increased groundwater elevations, while reduced precipitation periods correspond with declining groundwater elevations.

The NRC staff also reviewed the applicant's evaluation of West Coloma Creek interactions with the A Sand. The applicant identified that temporary changes in creek stage following precipitation events may influence local hydraulic gradients; however, such effects are expected to be localized and short duration.

The NRC staff finds that the applicant appropriately considered seasonal and short-term variations in A Sand groundwater conditions. The NRC staff concludes that the observed fluctuations do not significantly affect the overall groundwater flow direction or groundwater transport evaluation.

C Sand Temporal Trends

The NRC staff reviewed the applicant's evaluation of groundwater elevation trends within the C Sand. The applicant identified that C Sand groundwater elevations remained relatively stable during the monitoring period, with generally less than approximately 1 ft (0.3 meter) of variation in most monitoring wells.

The NRC staff reviewed the applicant's evaluation of C Sand potentiometric surface maps and hydraulic gradients over the monitoring period. The NRC staff finds that groundwater flow direction and hydraulic gradients within the C Sand remained consistent.

The NRC staff concludes that the temporal groundwater data supports the applicant's interpretation of stable C Sand groundwater flow conditions.

E Sand Temporal Trends

The NRC staff reviewed the applicant's evaluation of groundwater elevation trends within the E Sand. The applicant identified that E Sand groundwater elevations were generally stable, with limited variability during the monitoring period.

The NRC staff considered the applicant's evaluation of isolated groundwater elevation variations and the potential influence of measurement conditions or localized effects. The NRC staff finds that these variations do not represent a significant change in groundwater flow behavior.

The NRC staff concludes that the applicant adequately evaluated temporal groundwater trends within the E Sand.

Seasonal Variability

The NRC staff reviewed the applicant's assessment of seasonal groundwater variations based on groundwater elevation data collected over multiple seasons. The applicant concluded that seasonal variations are most apparent in the A Sand and are less pronounced in the C and E Sand units.

The NRC staff finds that the applicant appropriately evaluated seasonal groundwater variability and demonstrated that seasonal changes do not significantly alter groundwater flow directions, hydraulic gradients, or groundwater transport assumptions used in the safety analysis.

Seasonal fluctuations in groundwater influence the hydraulic gradient in groundwater flow and also impact the transport mechanisms of radioactive liquid effluents. The revised analysis of accidental release of radioactive liquid effluents hinges on relaxation of the conservative assumptions used in the screening-level analysis. Therefore, adequate characterization of seasonality is important to both the groundwater and accidental release scenario analysis which requires demonstration of compliance with 10 CFR Part 20 effluent concertation limits (ECLs), at the OL application stage.

Aquifer Properties

Aquifer properties were determined using field testing methods including slug tests and aquifer pumping tests. Parameters evaluated include transmissivity, storativity, and hydraulic conductivity (K).

The NRC staff reviewed the applicant's evaluation of aquifer properties relevant to groundwater flow and subsurface transport assessments. The applicant evaluated hydraulic properties using site-specific field testing, including slug tests and aquifer pumping tests.

The NRC staff's review considered hydraulic conductivity, transmissivity, storativity, aquifer thickness, and effective groundwater flow characteristics. The NRC staff reviewed information provided in the PSAR, held discussions with the applicant, and audited supporting technical documentation.

The NRC staff finds that the applicant has adequately characterized the hydraulic properties of the groundwater-bearing units beneath the LMGS site.

Hydrogeological Parameters

The NRC staff reviewed the applicant's evaluation of hydrogeological parameters obtained from slug testing and aquifer pumping tests.

Hydraulic Conductivity Evaluation

The applicant performed slug testing in site monitoring wells to estimate hydraulic conductivity values for the A, C, and E Sand units. The applicant analyzed slug test results using appropriate analytical methods, including the Hvorslev and Bouwer-Rice methods, based on well completion conditions.

The NRC staff reviewed the distribution of hydraulic conductivity values and the applicant's interpretation that variations reflect heterogeneous sedimentary conditions, including differences in sand, silt, and clay content.
The NRC staff finds that the applicant appropriately evaluated hydraulic conductivity using site-specific measurements and appropriately considered spatial variability within the water-bearing units.

Aquifer Pumping Tests

The NRC staff reviewed the applicant's aquifer pumping tests conducted in the C and E Sand units. The applicant performed pumping tests using dedicated test wells and monitoring wells to evaluate transmissivity, storativity, and hydraulic conductivity.

The NRC staff reviewed the applicant's evaluation that pumping test results did not indicate significant hydraulic communication between the A, C, and E Sand units. Specifically, the NRC staff reviewed monitoring data showing no measurable drawdown response in overlying or adjacent aquifer units during pumping activities.

The NRC staff finds that the aquifer pumping tests provide additional support for the applicant's hydrogeologic conceptual model and conclusion that the groundwater-bearing zones are hydraulically separated.

Groundwater Velocity

The NRC staff reviewed the applicant's groundwater velocity calculations based on measured hydraulic gradients, hydraulic conductivity values, and assumed effective porosity values.

The NRC staff finds that the applicant appropriately calculated groundwater velocities for use in the groundwater transport evaluation. The calculated velocities indicate relatively slow groundwater movement through the LMGS site hydrostratigraphic units.

The NRC staff concludes that the applicant's evaluation of hydrogeological parameters provides an acceptable basis for groundwater pathway assessments.

Geotechnical Parameters

The NRC staff reviewed the applicant's discussion of geotechnical parameters supporting the hydrogeologic characterization of the LMGS site.

The applicant stated that geotechnical testing results will be incorporated as additional information becomes available and that currently available geotechnical information used in the groundwater evaluation consists primarily of grain-size analyses and lithologic characterization from subsurface investigations.

The NRC staff reviewed the applicant's use of grain-size information to refine lithologic interpretations, hydrostratigraphic cross sections, and the conceptual LMGS site model.

The NRC staff finds that the applicant has appropriately incorporated available geotechnical information into the groundwater evaluation. The NRC staff concludes that additional geotechnical information, when available, should be evaluated during subsequent licensing stages to confirm consistency with the groundwater conceptual model and final design assumptions.

Hydrogeochemical Characteristics

The NRC staff reviewed the applicant's evaluation of hydrogeochemical characteristics relevant to groundwater conditions at and surrounding the LMGS site. The applicant evaluated available regional groundwater quality information obtained from TWDB databases and monitoring wells located within approximately 6 mi (9.6 km) of the LMGS site.

The NRC staff reviewed available analytical data for selected inorganic constituents and radionuclides, including reported concentrations of radium-228, uranium, alkalinity, and nitrate.

The NRC staff finds that the applicant appropriately characterized available regional hydrogeochemical conditions. The NRC staff notes that groundwater is not used as an operational water source at the LMGS site and that the available hydrogeochemical data provides an appropriate baseline for understanding regional groundwater conditions.

The NRC staff concludes that the applicant has provided sufficient information regarding hydrogeochemical characteristics to support the groundwater evaluation. The NRC staff also finds that the proposed groundwater monitoring program described in PSAR section 2.4.12.4 provides an appropriate mechanism for obtaining additional site-specific groundwater quality data during construction and operation.

2.4.12.1.3 Subsurface Pathways

The NRC staff reviewed the applicant's evaluation of subsurface pathways associated with potential groundwater transport from the LMGS site. The NRC staff's review considered the applicant's hydrogeologic conceptual site model (CSM), groundwater flow model development, numerical modeling approach, model calibration, and the application of groundwater modeling results to support the evaluation of potential radionuclide transport pathways.

The NRC staff's evaluation was based on information provided in the PSAR, audit discussions with the applicant's groundwater modeling and hydrogeology subject matter experts, and supplemental technical information and study reports provided through the electronic reading room.

The NRC staff evaluated whether the applicant adequately characterized the subsurface groundwater system and whether the groundwater model provides an appropriate technical basis for estimating groundwater travel times associated with postulated releases. The NRC staff finds that the applicant provided an acceptable evaluation of subsurface pathways relevant to the LMGS safety analysis.

Groundwater Flow Model

The NRC staff reviewed the applicant's development and application of a groundwater flow model to evaluate groundwater movement and potential subsurface transport pathways beneath the LMGS site.

The applicant developed a numerical groundwater flow model to simulate groundwater conditions within the A Sand unit of the Chicot Aquifer. The applicant identified the A Sand as the groundwater-bearing zone most likely to be affected by a potential accidental release because of its shallow occurrence, proximity to the proposed facility, and potential connection to a groundwater discharge location.

The NRC staff reviewed the applicant's basis for modeling only the A Sand and considered the results of site characterization activities, including hydraulic testing, groundwater monitoring, and aquifer pumping tests. The NRC staff reviewed the applicant's conclusion that the A, C, and E Sand units are not significantly hydraulically connected based on observed groundwater elevations and the absence of measurable drawdown response between aquifer units during pumping tests.

The NRC staff finds that the applicant's decision to model only the A Sand is technically justified for the intended purpose of evaluating potential groundwater transport pathways associated with a postulated accidental release.

The applicant developed the groundwater flow model using the U.S. Geological Survey (USGS) Model Muse groundwater modeling environment with MODFLOW as the groundwater flow simulation code and MODPATH for particle tracking analyses.

The NRC staff reviewed the applicant's modeling approach, including model objectives, conceptualization, boundary conditions, hydraulic parameters, calibration approach, and intended use of the model. The NRC staff finds that the groundwater flow model is appropriately developed for estimating groundwater flow paths and travel times for the LMGS groundwater pathway assessment.

Hydrogeologic Conceptual Site Model

The NRC staff reviewed the description of the hydrogeologic CSM developed for the LMGS site. The CSM provides a qualitative understanding of how local and regional hydrogeologic conditions affect groundwater flow beneath the LMGS site. The primary aquifer system is the Chicot Aquifer, but within the top 200 ft (61 m) of the LMGS subsurface, three key sand units: A Sand, C Sand, and E Sand, form distinct water-bearing zones, each separated by low-permeability clay aquitards.

- A Sand is the shallowest and thinnest (avg. ~4 ft (1.2 m) thick), composed of silty or clayey sands with low hydraulic conductivity (K) and low gradients, confined above and below by ~13 ft (4.0 m) and ~55 ft (17 m) clay aquitards respectively. It is most likely to be affected by potential accidental releases.

- C Sand is thicker (~29 ft (8.8 m) avg.), has more consistent lateral flow, and is separated from E Sand by a ~54 ft (16 m) thick clay unit.

- E Sand is deeper (encountered at 18–118 ft (5.5 36 m) below the C Sand), with flow toward the Gulf of Mexico [*sic*]. Thickness may be underestimated as most borings terminated at 200 ft (61.0 m) depth.

The applicant's CSM integrates information from:

- regional hydrogeology of the Gulf Coast Aquifer system;
- site-specific subsurface investigations;
- hydrostratigraphic interpretations;
- groundwater elevation monitoring;
- hydraulic testing results;
- surface water features; and
- groundwater use information.

The NRC staff finds that the applicant appropriately incorporated available site characterization data into the CSM.

Hydrostratigraphic Framework

The NRC staff reviewed the applicant's characterization of the subsurface groundwater system as consisting of three principal water-bearing zones:

- A Sand;
- C Sand; and
- E Sand.

The applicant identified intervening clay and silt units that act as aquitards separating the water-bearing zones.

The NRC staff reviewed the applicant's interpretation that the A Sand is the primary groundwater pathway of interest for potential accidental release evaluations. The NRC staff considered the applicant's evaluation of the thickness, hydraulic conductivity, and continuity of the A Sand unit.

The NRC staff finds that the applicant adequately characterized the hydrostratigraphic framework beneath the LMGS site.

Groundwater Flow Conditions

The NRC staff reviewed the applicant's evaluation of groundwater flow conditions within each water-bearing zone.

The applicant identified the following general groundwater flow patterns:

- A Sand groundwater flow, generally toward the east and northeast, with localized convergence toward West Coloma Creek;

- C Sand groundwater flows, generally toward the west; and

- E Sand groundwater flows, generally toward the south and southeast.

The NRC staff reviewed the applicant's evaluation of groundwater elevations and hydraulic gradients supporting these interpretations.

The NRC staff finds that the applicant's groundwater flow interpretations are consistent with available site-specific monitoring data and regional hydrogeologic conditions.

Hydraulic Separation of Water-Bearing Zones

The NRC staff reviewed the applicant's evaluation of hydraulic separation between the A, C, and E Sand units.

The NRC staff considered:

- differences in groundwater potentiometric elevations;
- thickness and properties of intervening clay units;

- aquifer pumping test results; and
- absence of measurable hydraulic response between monitored aquifer zones.

The NRC staff finds that the applicant has adequately demonstrated limited hydraulic communication among the water-bearing zones evaluated.

Groundwater Sources and Sinks

The NRC staff reviewed the applicant's identification of groundwater sources and sinks incorporated into the CSM.

The applicant identified:

- regional recharge areas associated with the Gulf Coast Aquifer system;
- potential localized recharge influences from Cooling Water Basin No. 5; and
- West Coloma Creek as a potential groundwater discharge location.

The NRC staff finds that the applicant appropriately incorporated relevant groundwater boundaries and hydrologic features into the conceptual model.

Seasonal Considerations

The NRC staff reviewed the applicant's evaluation of seasonal groundwater variability and its potential effect on groundwater flow conditions.

The applicant identified that seasonal fluctuations are most pronounced in the A Sand and are associated primarily with precipitation patterns and localized surface water interactions. The applicant concluded that temporary changes in groundwater gradients do not significantly affect long-term groundwater transport conditions.

The NRC staff reviewed the applicant's evaluation and finds that seasonal groundwater variations have been appropriately considered in the groundwater pathway assessment.

Groundwater Model Development

The NRC staff review of the description of the groundwater model confirmed only the A sand was included in the model. The decision is based on:

- Its proximity to the surface,
- Its susceptibility to releases,
- Demonstrated lack of hydraulic connectivity to the C and E Sands.

A three-dimensional, one-layer model with variable thickness was developed using site-specific hydrostratigraphy and groundwater data from site investigations and regional sources. A Bouwer-Rice geometric mean K of 1.43 ft/day (0.436 m/day) was used as the initial horizontal hydraulic conductivity across the domain.

Four hydrostratigraphic cross sections (A-A′ through D-D) supported interpolation of strata elevations and definition of model boundaries.

Numerical Model

The NRC staff reviewed the applicant's numerical groundwater flow model formulation, including the model domain, grid configuration, layer representation, solver selection, and simulation assumptions.

The applicant developed a three-dimensional, variable-thickness, one-layer groundwater flow model to represent groundwater movement within the A Sand beneath and surrounding the LMGS site. The model domain was selected to include the proposed LMGS site and surrounding hydrogeologic features considered relevant to groundwater flow conditions. The NRC staff reviewed the applicant's model grid configuration consisting of 181 columns, 132 rows, and one active model layer with a uniform grid spacing of 50 ft by 50 ft (15 m by 15 m). The applicant represented the A Sand as a confined aquifer unit with variable thickness based on interpreted hydrostratigraphic elevations derived from site investigation data.

The NRC staff reviewed the applicant's use of MODFLOW to simulate groundwater flow and MODPATH to perform particle tracking analyses for estimating groundwater travel times from a postulated release location to the nearest potential groundwater discharge location.

The NRC staff finds that the applicant selected an appropriate numerical modeling approach for the intended application. The model incorporates the relevant hydrogeologic features identified through site characterization and provides an acceptable basis for evaluating groundwater flow paths and potential transport times.

The NRC staff notes that the model was developed for the specific purpose of evaluating groundwater travel times associated with potential accidental release scenarios. The NRC staff finds that the applicant appropriately identified the limitations of the model and that additional evaluation would be necessary before applying the model for purposes beyond those described in the PSAR.

Boundary Conditions

The NRC staff reviewed the applicant's selection and implementation of groundwater model boundary conditions.

The applicant identified that the A Sand is a confined groundwater unit, so the applicant did not include direct recharge and evapotranspiration in the model. The applicant incorporated groundwater boundaries representing hydraulic influences identified during development of the hydrogeologic CSM.

The NRC staff reviewed the following model boundary conditions:

- A constant head boundary representing Cooling Water Basin No. 5;
- Northern and western constant head boundaries representing groundwater elevations interpreted from potentiometric surface maps; and
- Drain boundaries representing groundwater discharge to West Coloma Creek.

The NRC staff reviewed the applicant's representation of Cooling Water Basin No. 5 as a constant head boundary. The applicant assumed that the basin may provide a localized

hydraulic influence on the A Sand based on available groundwater elevation data and previous evaluations of basin leakage.

The NRC staff reviewed the representation of West Coloma Creek as a groundwater discharge feature. The applicant assigned drain boundaries to simulate groundwater discharge consistent with the interpretation that the creek may serve as a local groundwater sink.

The NRC staff finds that the applicant appropriately incorporated relevant groundwater sources and sinks into the numerical model. The selected boundary conditions are consistent with the applicant's hydrogeologic CSM and available groundwater monitoring data.

Model Calibration

The NRC staff reviewed the applicant's model calibration methodology, calibration targets, parameter adjustments, and calibration results.

The applicant calibrated the groundwater model by adjusting uncertain parameters, including:

- hydraulic conductivity;
- drain leakage characteristics; and
- constant head boundary locations and elevations.

The applicant evaluated calibration performance by comparing simulated groundwater elevations with observed groundwater elevations collected during the site investigation period.

The NRC staff reviewed the applicant's calibration criteria, which included evaluation of average residual differences between simulated and observed groundwater elevations and overall model water balance performance.

The NRC staff reviewed the applicant's calibration results and supporting information provided during audit discussions and through supplemental documentation. The NRC staff finds that the calibrated model reasonably represents average groundwater elevation conditions observed at the LMGS site during the monitoring period.

The NRC staff recognizes that the groundwater model represents a static, steady-state evaluation and is intended to estimate groundwater travel times under representative groundwater conditions. The NRC staff finds that the applicant appropriately identified the intended application and limitations of the model.

The NRC staff concludes that the groundwater flow model calibration is acceptable for supporting the groundwater pathway evaluation.

Groundwater Modeling Summary and Conclusions

The NRC staff reviewed the applicant's groundwater modeling summary and conclusions regarding groundwater flow conditions and potential subsurface transport pathways at the LMGS site.

The applicant developed and calibrated a three-dimensional, variable-thickness, one-layer groundwater flow model representing the A Sand unit of the Chicot Aquifer. The model was developed to evaluate groundwater movement and support estimation of groundwater travel times associated with a postulated accidental release.

The NRC staff reviewed the applicant's modeling approach, including:

- selection of the modeled hydrostratigraphic unit;
- representation of groundwater flow boundaries;
- hydraulic parameter selection;
- model calibration;
- groundwater flow simulations; and
- intended use of particle tracking results.

The NRC staff finds that the applicant appropriately identified the A Sand as the groundwater unit most relevant to potential groundwater transport from the proposed LMGS facility. The NRC staff finds that the applicant's decision not to model the C and E Sand units is supported by site-specific hydraulic testing and groundwater monitoring data demonstrating limited hydraulic communication between the water-bearing zones.

The NRC staff reviewed the applicant's groundwater flow model results and finds that the model adequately represents the general groundwater flow conditions beneath the LMGS site. The model results are consistent with observed groundwater elevations, groundwater flow directions, and the applicant's hydrogeologic CSM.

The NRC staff finds that the groundwater flow model is acceptable for estimating groundwater travel times associated with potential accidental releases. The NRC staff notes that the model is intended for the specific application described in the PSAR and is not intended to predict groundwater conditions unrelated to the evaluated safety analysis without additional review.

Based on the information provided in the PSAR, and supplemental technical information and study reports provided during audit activities, the NRC staff concludes that the applicant has adequately evaluated subsurface groundwater pathways for the LMGS site.

The NRC staff concludes that the groundwater flow model and associated subsurface pathway evaluation provide an acceptable technical basis for assessing potential groundwater transport of radionuclides associated with postulated releases and support the conclusions presented in the LMGS safety analysis.

2.4.12.1.4 Monitoring or Safeguards Requirements

The NRC staff reviewed PSAR section 2.4.12.4, "Monitoring or Safeguards Requirements," regarding groundwater monitoring and safeguards at the LMGS site. The section describes the evaluation of the existing groundwater monitoring well network in accordance with NEI 07-07 and 10 CFR 20.1406, including consideration of well viability and the potential need for replacement wells following construction activities.

The proposed monitoring program includes periodic groundwater level measurements and geochemical sampling within the Chicot Aquifer during construction and following plant startup.

Monitoring is intended to identify changes in groundwater flow patterns and temporal groundwater level trends relevant to accident analysis and structural stability. The specific wells, sampling parameters, monitoring frequency, and duration will be established during detailed design and the OL application stage. Operational process and effluent monitoring will also be addressed at the OL application stage.

The PSAR also identifies safeguards to minimize potential adverse impacts to groundwater, including lined containment for applicable storage tanks and hazardous-material storage areas, emergency cleanup procedures, and other measures as necessary to prevent or minimize groundwater contamination.

Based on its review, the NRC staff finds that the information provided in PSAR section 2.4.12.4 adequately describes the proposed groundwater monitoring approach and safeguards and is acceptable.

2.4.12.2 Conclusions

The NRC staff evaluated the applicant's characterization of the LMGS site hydrogeologic system and its assessment of groundwater as a potential subsurface pathway using information in the PSAR, audit discussions, and supplemental technical information and studies provided in the eRR. The NRC staff's review considered the regional and site-specific hydrogeologic setting, groundwater use, groundwater flow and aquifer properties, hydrogeochemical conditions, groundwater modeling, and proposed monitoring and safeguards programs.

The applicant characterized groundwater beneath the site as part of the Gulf Coast Aquifer system, with the A, C, and E Sand units comprising the principal water-bearing zones. Low-permeability clay and silt intervals limit hydraulic communication among these units. Site characterization, hydraulic testing, and groundwater monitoring indicate generally low hydraulic gradients and groundwater velocities. The NRC staff finds that the applicant adequately characterized groundwater use, flow directions, hydraulic properties, seasonal variability, and potential subsurface pathways. The NRC staff also finds that the A Sand is appropriately represented as the primary groundwater pathway in the groundwater flow model and that the model and particle tracking analyses provide an acceptable basis for evaluating groundwater travel times and potential radionuclide transport.

The NRC staff further finds that the proposed groundwater monitoring and safeguards programs provide an appropriate framework for identifying and mitigating potential changes in groundwater conditions during construction and operation.

Based on its review, the NRC staff concludes that the applicant adequately characterized the groundwater system and appropriately evaluated groundwater as a potential pathway for offsite exposure based on the information currently available. The NRC staff determined that the groundwater evaluation in PSAR section 2.4.12 is acceptable and provides reasonable assurance that groundwater conditions and potential subsurface pathways have been appropriately considered in the design and SE of the proposed facility.

2.4.13 Accidental Releases of Radioactive Liquid Effluents into Surface Water and Groundwater

2.4.13.1 Technical Evaluation

The NRC staff reviewed the applicant's evaluation of the site's hydrogeological characteristics to determine whether the information provided describes the effects of accidental releases of radioactive liquid effluents in ground and surface waters on existing uses and known and likely future uses of ground and surface water resources.

The source term from a postulated accidental release is reviewed under SRP 11.2 following the guidance provided in Branch Technical Position (BTP) 11-6, "Postulated Radioactive Releases Due to Liquid containing Tank Failures". The source term is determined from a postulated release from a single tank outside of containment. Normal and accidental releases are also considered in the applicant's ER.

The NRC staff reviewed the evaluation of the hydrogeological characteristics of the LMGS site. The NRC staff reviewed the PSAR to determine whether the evaluation assesses the effects of accidental releases of radioactive liquid effluents in ground and surface waters on existing uses and known and likely future uses of ground and surface water resources.

The specific areas of review are the following:

1) Alternate Conceptual Models: The NRC staff reviewed alternate conceptual models of the hydrology at the LMGS site that reasonably bound hydrogeological conditions at the site in as much as these conditions affect transport of radioactive liquid effluent in the ground and surface water environment. The NRC staff review examined whether the alternate conceptual models present a more realistic representation of the geometry of the hydrologic features and boundary conditions.

2) Pathways: The NRC staff reviewed the bounding set of plausible surface and subsurface pathways from potential points of accidental release to determine the critical pathways that may result in the most severe impact on existing uses and known and likely future uses of ground and surface water resources in the vicinity of the LMGS site.

3) Characteristics that Affect Transport: The NRC staff reviewed the ability of ground and surface water environment with respect to their ability to delay, disperse, dilute, or concentrate accidentally released radioactive liquid effluent during its transport. The NRC staff's review includes assessment of scenarios wherein accidental release of radioactive effluents is combined with hydrologic extreme events such as floods or low flows.

4) Consideration of Other Site-Related Evaluation Criteria: The NRC staff review includes assessment of scenarios wherein accidental release of radioactive effluents is combined with potential effects of seismic and non-seismic events (e.g., assessing effects of hydraulic structures located upstream and downstream of the plant in the event of structural or operational failures and the ensuing sudden changes in the regime of flow).

The NRC staff reviewed the evaluation performed by the applicant to assess hydrogeologic characteristics that affect the accidental releases of radioactive liquid effluents in ground and surface waters on existing uses and known and likely future uses of ground and surface water resources. The NRC staff also reviewed information on how the construction of the plant would influence the natural pathways for radionuclide transport.

The NRC staff reviewed the applicant's evaluation of hypothetical accidental releases of radioactive liquid effluents to groundwater and surface water, as described in PSAR section 2.4.13. The NRC staff's review included the information provided in the PSAR, supporting technical reports and calculations available in audit discussions with the applicant's hydrogeologic, geotechnical, and radiological subject matter experts to clarify technical assumptions, analytical methods, and regulatory considerations.

The applicant evaluated a bounding accidental release involving an instantaneous release of 75 cubic meters (75,000 liters) of radioactive liquid waste within the NI. The applicant assumed that the released liquid enters the A Sand water-bearing zone and migrates toward West Coloma Creek. For the initial evaluation, the applicant conservatively assumed instantaneous migration into groundwater without credit for infiltration through the vadose zone, dilution, hydrodynamic dispersion, radioactive decay, or sorptive retardation. The applicant compared the initial radionuclide concentrations with the ECLs specified in 10 CFR Part 20, Appendix B, table 2, and demonstrated that the initial source term exceeds the applicable NRC limits.

The applicant subsequently evaluated groundwater transport and radionuclide attenuation to determine whether radionuclide concentrations at potential receptor locations would remain below applicable regulatory limits following subsurface transport. The NRC staff reviewed the applicant's overall analytical methodology, including the initial seepage velocity assessment, groundwater particle tracking analysis, sensitivity evaluations, and radionuclide attenuation analyses. The NRC staff also reviewed the assumptions regarding the effects of engineered site grading and placement of fill material on groundwater flow and contaminant transport.

The NRC staff finds that the applicant's overall analytical framework is because it applies a tiered approach that progresses from conservative screening calculations to increasingly realistic analyses incorporating groundwater flow modeling and radionuclide attenuation mechanisms. The NRC staff's evaluation of the individual analytical components is discussed in the following subsections.

2.4.13.1.1 Initial Assessment Using Seepage Velocities

The applicant performed an initial screening evaluation using Darcy-based seepage velocity calculations to estimate the travel time required for radionuclides released into the A Sand water-bearing zone to migrate toward West Coloma Creek. The applicant estimated average and maximum groundwater seepage velocities using representative hydraulic conductivity, hydraulic gradient, and effective porosity values developed during the site hydrogeologic characterization described in PSAR section 2.4.12. Based on these calculations, the applicant estimated conservative groundwater travel times ranging from approximately 5 to 40 years.

The NRC staff reviewed the seepage velocity calculations, including the hydraulic parameters used in the analysis and their consistency with the site characterization information presented in PSAR section 2.4.12 and supporting hydrogeologic reports reviewed during the audit. The NRC

staff finds that the screening calculations appropriately provide an upper-bound estimate of groundwater travel time because they assume direct flow paths and do not account for aquifer heterogeneity, groundwater flow convergence, radionuclide retardation, radioactive decay, or dispersion. Consequently, the NRC staff finds that the seepage velocity assessment appropriately serves as a conservative screening analysis. To account for the actual transport of radionuclides, the applicant performed a detailed groundwater transport analysis based on the flow velocity and inventory of radionuclides.

2.4.13.1.2 Assessment Using Particle Tracking

To provide a more representative evaluation of groundwater transport, the applicant performed particle tracking analyses using the MODPATH code coupled with the site MODFLOW groundwater flow model described in PSAR section 2.4.12. The applicant simulated particle releases throughout the Nuclear and Conventional Island and tracked particle movement toward potential downgradient discharge locations over periods extending to 106 years.

The NRC staff reviewed the conceptual basis of the groundwater flow model, the implementation of the particle tracking simulations, the selected effective porosity, and the interpretation of the model results. The NRC staff also audited supplemental modeling documentation and discussed the development of the groundwater model and particle tracking simulations during multiple audit sessions.

The particle tracking analysis predicted that the earliest groundwater particles originating near the eastern portion of the NI would reach West Coloma Creek after approximately 11.6 years, whereas releases originating nearer the center or western portions of the site would require substantially longer travel times. The NRC staff finds that the particle tracking analysis provides a more technically representative estimate of groundwater migration than the initial seepage velocity calculations because it accounts for the simulated groundwater flow field and predicted flow paths.

The NRC staff further finds that the applicant conservatively evaluated unattenuated particle movement and appropriately deferred consideration of radionuclide attenuation until later stages of the analysis. Accordingly, the NRC staff concludes that the particle tracking analysis provides an acceptable basis for estimating radionuclide travel times for the purposes of evaluating hypothetical accidental releases.

2.4.13.1.3 Sensitivity Analysis

The applicant performed sensitivity analyses to evaluate the influence of potential release location and groundwater travel time on the predicted migration of radionuclides. Additional particle tracking simulations were conducted using multiple release locations distributed across the Nuclear and Conventional Island, and hypothetical travel times equal to one-half of the shortest calculated travel time were evaluated to assess the sensitivity of the radionuclide concentration calculations.

The NRC staff reviewed the sensitivity analyses and discussed the selected scenarios during audit discussions. The NRC staff finds that evaluating alternative release locations provides additional confidence that the groundwater flow model reasonably represents the range of potential transport pathways within the NI. The NRC staff also finds that the hypothetical one-

half travel-time evaluation provides an appropriately conservative sensitivity analysis demonstrating that moderate uncertainty in predicted groundwater travel time would not significantly alter the overall conclusions regarding radionuclide attenuation and compliance with the unity rule.

Accordingly, the NRC staff concludes that the applicant's sensitivity analyses reasonably demonstrate the robustness of the groundwater transport evaluation.

2.4.13.1.4 Attenuation Factors

The applicant evaluated attenuation of radionuclides during groundwater transport by considering radioactive decay and sorptive retardation within the A Sand water-bearing zone. These attenuation mechanisms were applied following completion of the conservative groundwater flow and particle tracking analyses.

Radioactive Decay

The applicant calculated radionuclide decay using isotope-specific half-lives obtained from established technical references and applied standard radioactive decay equations to determine the remaining radionuclide activity at the estimated groundwater travel times.

The NRC staff reviewed the applicant's methodology, decay equations, and referenced half-life data. The NRC staff finds that the applicant applied accepted radioactive decay relationships using appropriate isotope-specific decay constants. The NRC staff also finds that the applicant appropriately incorporated decay over the modeled groundwater travel times determined from the particle tracking analysis.

Advective Retardation (Sorption)

The applicant evaluated radionuclide retardation using linear equilibrium sorption relationships and conservative distribution coefficients (K_d values) representative of sandy aquifer materials. Retardation factors were calculated using accepted hydrogeologic relationships relating K_d values, bulk density, and effective porosity.

The NRC staff reviewed the applicant's selection of K_d values, calculation of retardation factors, and resulting radionuclide travel times. During audit discussions, the NRC staff requested clarification regarding the basis for selecting conservative sorption coefficients and the treatment of radionuclides lacking published K_d values. The applicant clarified that where published values were unavailable, no retardation was assumed, resulting in conservative transport predictions.

The NRC staff finds that this approach is appropriately conservative because the use of sandy aquifer K_d values generally minimizes retardation relative to finer-grained sediments, while assuming no retardation for radionuclides lacking published values further biases the analysis toward shorter travel times.

The applicant combined the effects of groundwater travel time, sorptive retardation, and radioactive decay to estimate radionuclide concentrations at the modeled discharge location. The NRC staff reviewed the revised estimates which account for the various attenuation factors.

The applicant demonstrated that by incorporating attenuation factors such as retardation and radioactive decay the ECL estimates at the postulated receptor point will decrease and satisfy the ECL limits in regulatory requirements. This demonstrates the viability of meeting the ECL limits in 10 CFR Part 20 for the LMGS site, consistent with 10 CFR 100.20(c)(3). The NRC staff finds that the applicant's postulated source term and radionuclide transport analysis reasonably demonstrate compliance with the ECL limits. Further revision to this analysis with actual Xe-100 source term inventories will be reviewed at the OL stage.

2.4.13.2 Conclusions

The NRC staff reviewed the applicant's evaluation of hypothetical accidental releases of radioactive liquid effluents using the information provided in the PSAR, supporting calculations and technical reports available during audit discussions with the applicant.

The NRC staff finds that the applicant used a conservative, stepwise analytical approach that progresses from screening-level seepage velocity calculations to groundwater flow modeling, particle tracking, sensitivity analyses, and radionuclide attenuation evaluations. The NRC staff concludes that the analytical methodology is technically acceptable and provides reasonable assurance that radionuclide transport resulting from the postulated accidental release would be sufficiently attenuated before reaching the modeled discharge location.

2.4.14 Technical Specifications and Emergency Operation Requirements

2.4.14.1 Technical Evaluation

The NRC staff evaluated the applicant's probable subjects of technical specifications (TS) in Part IV of the CP application and preliminary plans for emergency operating requirements associated with external flooding hazards described in the PSAR to determine whether the information provides a viable path for maintaining the capability of SSCs credited with performing safety functions during and following external flood events.

The applicant stated that the natural site grade is approximately 27 ft (8.2 m) NAVD 88 and that the finished plant floor elevation is established at elevation 31.5 ft (9.6 m) NAVD 88. The applicant further stated that the controlling hydrologic hazards are evaluated in PSAR sections 2.4.2 through 2.4.9 and that the flood protection requirements for SR and certain NSRST SSCs are established in PSAR section 2.4.10.

As discussed in section 2.4.10 of this SE, the NRC staff finds that the applicant's flood protection strategy relies primarily on passive, elevation-based flood protection, supplemented, where necessary, by passive flood barriers for SR and certain NSRST SSCs located below the design-basis stillwater flood elevation.

The applicant stated that TS and Emergency Operating Procedures (EOPs) are currently under development and will be provided as part of the FSAR and OL application. The applicant further stated that administrative controls and emergency procedures are not credited to demonstrate protection of SR SSCs from design-basis flood hazards but may be implemented to provide additional defense in depth or to support mitigation of beyond-design-basis conditions.

The SR reactor cooling and decay heat removal provided by the passive reactor cavity cooling system (RCCS) requires water addition approximately 72-hours after the start of the event.

However, this duration will be longer for a design-basis flooding event, where LME has indicated in PSAR section 6.1 that operating procedures will include shutting down the reactors prior to inundation reaching the site. This will reduce the heat load on RCCS if active systems are lost during the DBHL flooding event. Additional information on TS and EOPs for responding to flooding hazards will be reviewed at the OL stage, when available.

The NRC staff reviewed the applicant's description in PSAR section 2.4.14. The NRC staff considered the flood hazards evaluated in PSAR sections 2.4.2 through 2.4.11, including flooding due to storm surge, precipitation events, channel migration, low water conditions, and other external flooding mechanisms. The NRC staff's review included the information presented in PSAR section 2.4.14, supporting documentation available during audit discussions with the applicant regarding the overall flood protection strategy and the role of administrative and operational controls in protecting SR and certain NSRST SSCs.

2.4.14.2 Conclusions

The NRC staff reviewed the applicant's probable subjects of TS and emergency operating requirements associated with external flooding hazards at LMGS. The NRC staff finds that the information provides a viable path for maintaining the capability of SSCs credited with performing safety functions during and following external flood events.

Consistent with 10 CFR 50.34(b)(6), TS and EOPs for LMGS will be provided as part of the OL application.

2.5 Geology

PSAR section 2.5 discusses the geological, seismological, and geotechnical characteristics of the site for its suitability to support the construction and operation of a nuclear facility. This section assesses the site's geologic setting, seismic hazards, soil properties, and other relevant geotechnical features that could impact the safety of the LMGS site.

The applicable regulatory requirements related to the geologic description of the site are:

- 10 CFR 50.34(a)(1)(ii)(D), which requires that, among other things, site characteristics must comply with 10 CFR Part 100. Subpart B of 10 CFR Part 100 provides "Evaluation Factors for Stationary Power Reactor Site Applications on or After January 10, 1997.";

- 10 CFR Part 50, Appendix A, Section I, Criterion 2;

- 10 CFR Part 50, Appendix S;

- 10 CFR 100.21(d), which requires, in part, that the physical characteristics of the site, including geology, must be evaluated and site characteristics established such that potential threats from such physical characteristics will pose no undue risk to the type of facility proposed to be located at the site;

- 10 CFR 100.23(c), which requires, in part, that the geological characteristics of a site and its environs be investigated in sufficient scope and detail to permit an adequate evaluation of the proposed site; and

- 10 CFR 100.23(d), which requires, in part, that the geologic and seismic siting factors considered for design must include a determination of the potential for surface tectonic and nontectonic deformations and for other design conditions.

The applicable guidance for the evaluation of the geologic description of the site are:

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition." (ML070810350); and

- Regulatory Guide (RG) 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion" (ML070310619).

2.5.1 Basic Geologic and Seismic Information

Section 2.5.1 of this SE addresses PSAR section 2.5.1 of the LMGS CP application.

2.5.1.1 Technical Evaluation

The applicant stated that PSAR section 2.5.1 was developed in accordance with the guidance in NUREG-0800 and RG 1.208; therefore, the NRC staff reviewed PSAR section 2.5.1 for consistency with these referenced NRC guidance documents to determine that the basic geologic and seismic information meets regulations identified above. The NRC staff's objective was to determine whether the applicant provided adequate basic geologic and seismic information to support the evaluations in subsequent sections of the PSAR.

In PSAR section 2.5.1, the applicant presented geologic and seismic information at the regional (200-mile (322 km)), vicinity (25-mile (40 km)), area (5-mile (8 km)), and site scales. This information included the physiography, stratigraphy, lithology, geomorphology, tectonic setting, faulting and folding, and geologic history, with emphasis on the Quaternary period. The NRC staff reviewed this information and determined that the applicant characterized the geologic framework at each spatial scale with the level of detail expected for this stage of review.

Because the LMGS site is located approximately 20 mi (32 km) from the Exelon Victoria County Station (VCS), the applicant incorporated information from the VCS Early Site Permit (ESP) Application (Revision 1; ML12131A060, ML12131A063, ML12131A064) where the LMGS and VCS investigation areas overlap. The NRC staff evaluated this approach and finds the use of applicable VCS ESP information appropriate to support the applicant's regional and vicinity-scale descriptions because the VCS ESP geology siting requirements are similar to the LMGS geology siting requirements and because the VCS ESP investigation areas partially overlap with the LMGS regional and vicinity-scale investigation areas.

The NRC staff reviewed the applicant's description of the site's physiographic and geologic setting. The applicant characterized LMGS as lying within the nearly flat Coastal Prairies sub-province of the Gulf Coastal Plains, underlain by an approximately 30,000-foot-thick (9,144 meter-thick) wedge of post-rift sediments. The applicant identified the stratigraphic sequence from rift-graben basement through successive marine depositional cycles, including the Goliad Sand, Lissie Formation, and the Beaumont Formation at the surface. Because the applicant's description of the physiographic setting and the geology to surface aligns with commonly accepted geologic interpretations of the area (see references in SE section 2.5.1.4),

the NRC staff finds that the applicant provided an adequate characterization of the stratigraphy and its relevance to site conditions.

The applicant described site lithology as consisting of clays to sands deposited in a fluvial-deltaic environment. The description of site cross-section lithology in PSAR sections 2.5.1, 2.5.4, and 2.4.12 is not consistent; however, PSAR table 2.5.1-2 and the explanation in section 2.5.1.2.2 clarify that these different sections of the application employ different definitions of "sand." The NRC staff reviewed the applicant's use of the different definitions of "sand" and determined that the applicant adequately clarified the appropriate usage of the different definitions of "sand."

Regarding tectonic setting, the applicant explained that the site region reflects a long history of compressional forces associated with the Grenville, Ouachita, and Laramide orogenies, and extensional forces associated with Proterozoic rifting and the Mesozoic rifting which formed the Gulf of America (formerly known as Gulf of Mexico). The applicant stated that the modern coastal plain sedimentary wedge and associated faulting patterns result primarily from Mesozoic rifting, salt withdrawal, and crustal subsidence. The NRC staff determined that the applicant's characterization of the tectonic setting is adequate because it aligns with commonly accepted geologic interpretation.

The applicant identified the nearest basement-involved faults—the Balcones and Luling fault zones—as located outside the site vicinity and described the Texas Gulf Coast as a continental margin that has been tectonically stable during the Quaternary. The applicant reported no evidence of Quaternary growth faulting nor differential subsidence at the LMGS site. The NRC staff reviewed this information and determined that the applicant's conclusions regarding the absence of Quaternary tectonic deformation at the site are adequately supported.

The applicant evaluated dissolution and volcanic hazards and reported no soluble geologic units within the site area, no Quaternary volcanism within the site region, and no Quaternary volcanic deposits within the site vicinity. The NRC staff confirmed that these evaluations were based on credible geologic sources and were appropriately represented in the PSAR.

2.5.1.2 Proposed Permit Condition

The stability of the foundation can be impacted by the presence of growth faults. In PSAR subsection 2.5.3.2.2, the applicant states that growth faults have been mapped in the site vicinity and are interpreted to occur at depth in the LMGS site area. For evaluation of suitability of a proposed site, 10 CFR 100.23(c) requires that geologic data on tectonic and nontectonic surface deformation and site foundation material be obtained through review of pertinent literature and field investigations. 10 CFR 100.23(d)(2) explicitly states that geologic and seismic siting factors considered for design must include determination of the potential for tectonic and nontectonic surface deformation at the proposed site. 10 CFR 100.23(d)(4) explicitly states that siting factors for design considerations must include evaluation of the physical properties of the materials underlying the site. RG 1.208 specifically states that faults exposed in site excavations should be mapped and assessed in regard to rupture potential while walls and floors of the excavations are exposed, to include assessment of nontectonic surface deformation. Therefore, the NRC staff proposes the following geologic mapping permit condition:

Geologic Mapping Permit Condition: Long Mott Energy, LLC., (LME) shall perform geologic mapping of each foundation excavation for safety-related structures. For a mat resting on drilled deep foundation elements supporting safety-related structures, LME shall perform geologic mapping of exposed portions (where safely accessible) and shall perform geologic logging of soil cuttings obtained from portions constructed under slurry or temporary casing, or otherwise not accessible for direct observation. LME shall examine and evaluate the geologic features and conditions identified for consistency with the site characterization described in Preliminary Safety Analysis Report section 2.5. Once geologic mapping and logging information from the foundation excavation and deep foundation elements for safety-related structures are available for examination by the NRC staff, LME shall notify the Director of the Office of Advanced Reactors, or the Director's designee, as specified in Title 10 of the Code of Federal Regulations (10 CFR) Section 50.4, "Written communications."

2.5.1.3 Conclusion

Based on results of the NRC staff's review and evaluation of the detailed information presented in PSAR section 2.5.1, the NRC staff concludes that the applicant investigated, evaluated and established the geologic characteristics of the LMGS site region, site vicinity, site area and site location in accordance with 10 CFR 50.34(a)(1)(i), 10 CFR 50.34(a)(1)(ii)(D), 10 CFR Part 50, Appendix A, Section I, criterion 2, 10 CFR 100.21(d), 10 CFR 100.23(c), and 10 CFR 100.23(d) and consistent with guidance in NUREG-800 and RG 1.208.

2.5.2 Vibratory Ground Motion

The NRC staff reviewed PSAR section 2.5.2, which describes the applicant's ground motion assessment for LMGS. The applicant followed RG 1.208 and NUREG-0800 in its selection of the Central and Eastern United States Seismic Source Characterization (CEUS-SSC) and Next Generation Attenuation (NGA-East) models as the starting point for seismic hazard assessment at LMGS. Following RG 1.208, the applicant considered the adequacy of CEUS-SSC and NGA-East models for the site and included updates to the model to account for new information not available at the time these models were published.

The NRC staff reviewed PSAR section 2.5.2.5, which includes a description of the development of a site-specific, performance goal-based ground motion response spectra (GMRS), which involved two sets of site amplifications. The first set of site amplifications is based on data obtained from VCS and STP. The second set of site amplifications is based on site response analysis from geotechnical borehole and geophysical survey field investigations. The derivation of the first set of site amplification functions (SAFs) included assumptions that SAFs for LMGS would be the same shape and magnitude as SAFs from VCS and STP and would envelop site amplification developed from data gathered at the site. PSAR section 2.5.2.5.2 describes the validity of these assumptions.

The NRC staff reviewed PSAR sections 2.5.2.1 through 2.5.2.3, which describe the seismicity, geologic and tectonic characteristics of the site region, and correlation of earthquake activity with seismic sources. PSAR subsection 2.5.2.4 describes the probabilistic seismic hazard analysis (PSHA) and controlling earthquakes using approximate SAFs. PSAR subsection 2.5.2.5 describes seismic wave transmission of the site. PSAR subsection 2.5.6 describes the development of the GMRS using site-specific hazard using SAFs and associated uncertainties derived from site-specific data.

2.5.2.1 Seismicity

2.5.2.1.1 Technical Evaluation

CEUS-SSC Earthquake Catalog and Earthquake Catalog Update

The NRC staff reviewed PSAR sections 2.5.2.1.1 and 2.5.2.1.2, which describe the review and update of the earthquake catalog used to define the seismic sources for the LMGS site. The applicant starts with the CEUS-SSC catalog as the base model and updates the catalog following the procedures outlined in chapter 3 of the model documentation (EPRI, DOE, and NRC, 2012) when performing its catalog analyses. The applicant expanded the CEUS-SSC earthquake catalog to include earthquakes recorded from January 1, 2009, through May 31, 2024, for an area encompassing a 200 mi (320 km) radius from the site. These earthquakes were compiled from the following sources: USGS Advanced National Seismic System Comprehensive Earthquake Catalog; the Texas Seismological Network catalog; the Global Centroid Moment Tensor Solution catalog; and the catalog of relocated earthquakes developed by Li et al. (2021) in the Eagle Ford region. The applicant produced two versions of the final catalog: one with all human-induced earthquakes removed and one that retains the human-induced earthquakes.

The NRC staff finds the use of the CEUS-SSC (NUREG-2115) historical catalog as the base catalog supplemented by other established earthquake network catalogs and literature review adequately captures seismicity around the site. Extending the length of the historical catalog to May 31, 2024, is appropriate because it conforms to RG 1.208, which states that applicants should update seismic sources where there is significant new information.

Assessment of the Impact of the Updated Earthquake Catalog on Seismicity Rates

The NRC staff reviewed the one-sided Poisson test the applicant performed on the CEUS-SSC and updated catalogs and determined that the observed number of earthquakes without including human-induced events is consistent with the expected seismicity rates from the CEUS-SSC model. The applicant also performed this test on the number of observed earthquakes including human-induced events and their results showed a significant deviation from the expected rate from the CEUS-SSC model. The applicant addresses how they account for the impact of human-induced earthquakes to the seismic hazard of LMGS site in PSAR section 2.5.2.4.

The NRC staff finds the methodology and analysis of seismicity rates for a catalog retaining human-induced earthquakes and for a catalog without human-induced earthquakes for LMGS technically reasonable because it sufficiently accounts for all sources of ground motion that may impact the site.

Significant Earthquakes in the Site Region (200 Mi. [320 Km] Radius)

The NRC staff reviewed PSAR section 2.5.2.1.4 in which the applicant analyzed the concentration of seismicity within the Eagle Ford Shale play and lists four of the largest earthquakes reported within 200 mi (320 km) of the site: April 7, 2008, E[M] 4.84 Falls City, October2011, 2011 E[M] 4.78 Fashing, and February 17, 2024, E[M] 4.74 and E[M] 4.3 Falls City earthquakes.

The NRC staff performed an independent check of the historical seismicity from established earthquake data sources and determined the applicant's list of largest earthquakes to occur within a 200 mi (320 km) radius of the site is complete and, therefore, acceptable.

Seismicity Associated with Petroleum Industry

The applicant summarizes several studies linking earthquakes in the Eagle Ford region to petroleum industry activities and identifies the closest wells as being located over 80 km away from the site. The larger of the moderate magnitude earthquakes, such as the October 20, 2011, E[M] 4.78 Fashing event, produced noticeable ground shaking locally. The NRC staff finds it appropriate for the applicant to consider human-induced seismicity associated with petroleum industry activities due to the nearby location of the Eagle Ford Shale.

2.5.2.1.2 **Conclusion**

As required by 10 CFR 100.23(c), the NRC staff determined that the applicant investigated the seismological characteristics of the site in sufficient scope and detail. As discussed in RG 1.208, the NRC staff reviewed the use of and the modifications made to the source parameters of CEUS-SSC (NUREG-2115) to ensure that the applicant took considerable care to correct for incompleteness and to model the uncertainty in the rate of earthquake recurrence.

The NRC staff concludes that PSAR section 2.5.2.1 describing seismicity for LMGS is acceptable based on (1) the appropriate historical catalog length and source, (2) the completeness of the catalog with minimum magnitude greater than or equal to E[M] 2.9, and (3) the thorough analysis of seismicity rates, including taking into consideration uncertainty of rates due to human-induced earthquakes.

2.5.2.2 Geologic and Tectonic Characteristic of the Site Region

2.5.2.2.1 **Technical Evaluation**

The NRC staff reviewed PSAR section 2.5.2.2, which summarizes the source models used in the CEUS-SSC model that will be applied to the PSHA for LMGS. The summary includes repeated large magnitude earthquake sources (RLME) and background distributed seismicity sources zones, maximum earthquake magnitude (M_{max}) zones, and seismotectonic zones. Further details on the adjustments made to the model are discussed in PSAR section 2.5.2.4.

2.5.2.2.2 **Conclusion**

The NRC staff determined that the use of NRC-endorsed CEUS-SSC (NUREG-2115) and its model parameters as a starting point is acceptable for LMGS because it is consistent with RG 1.208 and therefore meets the requirement of 10 CFR 100.23(c) to investigate the seismological characteristics of the site in sufficient scope and detail to permit an adequate evaluation of the proposed site and its seismic hazards.

2.5.2.3 Correlation of Earthquake Activity with Seismic Sources

2.5.2.3.1 Technical Evaluation

The NRC staff reviewed the applicant's summary of two studies, Li et al. (2021) and McKeighan et al. (2022), correlating earthquakes to the Eagle Ford Shale and potential for future seismic activity. Results from Li et al. (2021) suggest the existence of unmapped faults in the Eagle Ford Shale and the study by McKeighan et al. (2022) show that small increases in pore pressure may trigger earthquakes in the region.

2.5.2.3.2 Conclusion

The NRC staff finds that the literature review addresses new information and conforms to RG 1.208. Therefore, the NRC staff finds that the literature review performed by the applicant to investigate seismological characteristics of a site is of sufficient scope and detail to meet the requirements in 10 CFR 100.23(c).

2.5.2.4 Probabilistic Seismic Hazard Analysis and Controlling Earthquakes

2.5.2.4.1 Technical Evaluation

The NRC staff reviewed the applicant's approach to performing the PSHA for LMGS, using CEUS-SSC (NUREG_2115) as a starting point, with modifications and new information included as outlined within the sections below. The applicant used NGA-East Ground Motion Model (NGA-East GMM) and made adjustments to the model for site location.

New Information Relative to Seismic Sources

The NRC staff reviewed the applicant's evaluation of new seismic information and updates considered for the CEUS-SSC model for the LMGS site, including potential changes to earthquake occurrence rate, maximum magnitude, and the addition of seismic sources in the site region.

The applicant made no changes to the CEUS-SSC model recurrence parameters and maximum magnitude distributions. This decision was based on the earthquake occurrence rate analysis of the historical catalog without human-induced earthquakes and from the analysis of the region's largest earthquakes, which are still smaller than the smallest maximum magnitudes modeled in CEUS-SSC.

The NRC staff reviewed the changes the applicant made to the Gulf Highly Extended Crust Zone (GHEX) with increased weight for northeast-southwest fault orientations based on studies of the region. The applicant chose not to incorporate the Saline River fault zone or the USGS 2023 National Seismic Hazard Model updates to the New Madrid fault source or Meers faults, stating that the distance from the site, the small magnitude or minor nature of the geometric changes, and associated uncertainties would not produce a significant impact on site hazard.

The NRC staff performed a high-level review of the parameter changes to GHEX and determined that the technical basis for this change is reasonable and supported by the applicant's literature review of the regional studies demonstrating northeast-southwest fault

rupture orientations are appropriate for the region. The NRC staff also reviewed the NGA-East GMM adjustment factors applied for LMGS and finds the technical basis for these changes to be reasonable based on site-specific considerations showing that these factors are not applicable to the tectonic setting of the site.

Eagle Ford Faulted Zone

The NRC staff reviewed the applicant's inclusion of the Eagle Ford Faulted Zone (EFFZ) to account for potential earthquakes that may be larger than what is currently observed. The applicant based this decision on its literature review studying the relationship between human activity and fault rupture in this region and a literature review on petroleum activity regulatory schemes. The potential for moderate to larger magnitude earthquakes would be captured by either the EFFZ or already modeled by CEUS-SSC distributed seismicity source zones. The EFFZ geometry and parameters are primarily based on McKeighan et al. (2022).

The NRC staff finds the inclusion of new source zone EFFZ reasonable and technically justified based on the applicant's analysis of the regional seismicity, which shows that incorporating petroleum industry-related earthquakes results in a higher seismicity rate and that the EFFZ is the primary source for the region's largest earthquakes that may impact the site. The NRC staff also determined the parameters defined for the EFFZ source zone, including the spatial geometry, spatial distribution of seismicity, M_{max}, fault orientation, and other associated model input parameters are technically justified and reasonable for LMGS based on the applicant's regional seismicity analysis and the supporting literature review used to establish the model inputs.

NGA-East Gound Motion Model

The NRC staff reviewed the applicant's use of NGA-East GMM and adjustments to the attenuation model for the GCR zone, inclusion of source depth adjustment, and not including a hanging wall effect adjustment. The NRC staff finds the justification provided by the applicant for these adjustments technically justified because they account for higher attenuation in the GCR and reflect the shallow earthquake depths characteristic of the area, and because hanging-wall effects are not applicable to the site.

Probabilistic Seismic Hazard Analysis for Reference Rock and Deaggregation

The NRC staff reviewed the applicant's PSHA results for the hazard at reference rock and the deaggregation of the hazard using the inputs and modifications as described earlier. The PSHA results show that for spectral frequencies of 2.5 Hz and higher and an annual frequency of exceedance (AFE) greater than 1E-5, the hazard is controlled by the EFFZ due to proximity to the site, potential for moderate to large magnitude earthquakes, and relatively high seismicity rates. At 1 Hz the contribution to the hazard from the three source types, the CEUS-SSC distributed seismicity sources, the RLME sources, and the EFFZ, are comparable. At low frequency motions such as 0.5 Hz, the dominant contributors to the hazard are RLME sources. At AFEs less than 1E-5, the distributed seismicity zones contribute the most to the hazard at all frequencies.

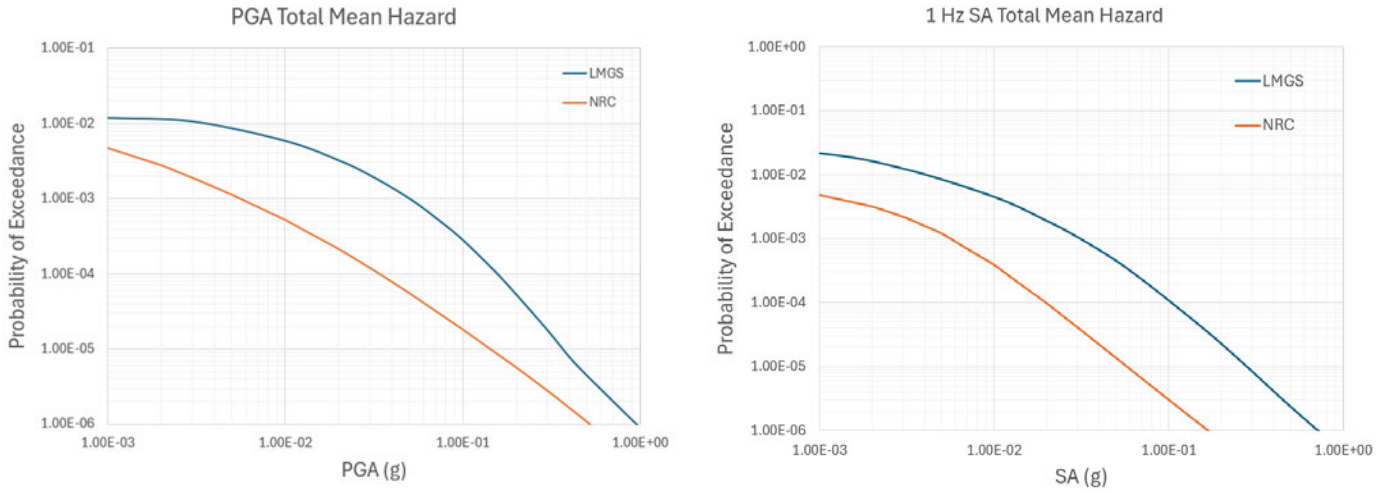
The NRC staff performed an independent PSHA using the standard CEUS-SSC inputs. A comparison between the PSHA performed by the NRC staff and the applicant's PSHA show that the applicant's analyses produce higher hazard compared to the standard CEUS-SSC inputs

across the entire range of acceleration values (Figure 2.5-1). The difference in hazard between the applicant's hazard results and the NRC staff's results is attributed mainly to the inclusion of EFFZ in the applicant's analyses.

2.5.2.4.2 **Conclusion**

Overall, NRC staff finds the applicant's PSHA model, the inputs and modifications, as well as the hazard results evaluation technically reasonable and supported by the applicant's analyses and literature review.

Figure 2.5-1 – PGA Total Mean Hazard and 1 Hz Total Mean Hazard



2.5.2.5 Seismic Wave Transmission Characteristics of the Site

2.5.2.5.1 **Technical Evaluation**

The NRC staff reviewed the applicant's approach to the development of the approximated SAFs and the development of the site-specific amplification functions.
The applicant used published data from nearby sites, VCS and STP, to derive approximate SAFs for the LMGS site. There were three main assumptions made in this process: the shape of the SAFs for LMGS would be the same shape as the SAFs from VCS and STP, the variation of SAFs for LMGS for the various levels of input ground motions at the reference rock horizon would be within the variation determined from published literature, and the assumed epistemic and aleatory uncertainty in site amplification would envelop the site-specific computed SAFs.

The NRC staff finds the approach to developing approximate SAFs technically reasonable based on the location of the referenced sites and the level of study performed at those nearby sites for the purpose of building a nuclear facility. As discussed below, the applicant also performed a site response analysis and demonstrated that the approximate SAFs are bounding the LMGS site-specific SAFs.

Approximate Site Amplification Functions

The NRC staff reviewed the applicant's approach to developing the approximate SAFs. The applicant summarizes the geologic conditions from a literature review of VCS ESP applications (Exelon, 2010; Exelon, 2012b) and the STP FSAR NINA COLA (2015b). Based on the literature review and preliminary boring logs at the LMGS site, the applicant derived an estimate of mean SAFs and uncertainties for LMGS through analysis of digitized VCS and STP SAFs. The applicant's literature review showed that VCS had lower amplification values than STP, but the shape of both amplification functions is similar. The applicant compares site amplification models for Central and Eastern North America (CENA) and GCR to median SAFs for VCS and STP and states that the similarities in shape of the median SAFs for VCS and STP are consistent with the referenced models.

The applicant applied three types of adjustments to the median SAFs for VCS and STP at the LMGS site. The first adjustment modified VCS SAFs to represent ground surface amplification instead of amplification at 100 ft (30.5 m) depth. The second adjustment was to STP SAFs to represent sediment thickness of 9 km instead of 10 km for LMGS. The third adjustment scaled VCS SAFs for AFE 1E-4 and 1E-5 to match STP reference rock peak ground acceleration (PGA) values and median SAFs. The applicant performed a sensitivity analysis to evaluate the effect of varying levels of reference rock PGA and the amplification patterns by the SAF in the referenced models. Epistemic uncertainty was based on referenced VCS and STP models (Exelon, 2012a; NINA, 2015a) and followed NRC recommendations (NRC, 2021b).

The NRC staff reviewed LME's methodology for developing approximate seismic wave transmission characteristics of the site. The applicant estimated site conditions for LMGS based on the site-specific investigations carried out at VCS and STP, preliminary boring logs at LMGS, and literature review. The NRC staff finds this methodology acceptable as the applicant adequately supports its analyses with data from nearby sites with comparable geologic conditions and preliminary site-specific information. The site amplification models and the evaluation of uncertainty were appropriate for the site region because they are supported by the regional geologic data and consistent with common practice for CENA PSHA. The NRC staff finds the use of approximate site characteristics acceptable because it is validated by the site-specific analyses and sufficiently envelopes the site-specific conditions.

Site Response Analysis

The NRC staff reviewed LME's methodology for developing site-specific seismic wave transmission characteristics of the site, including the input parameters, methodology, randomization for velocity profiles and material curves, calculation of broadened conditional mean spectra, and derivation of SAFs as discussed below.

Velocity Profiles

The NRC staff reviewed the applicant's derivation of the velocity profile, which was developed in two parts, the upper 655 ft (200 m) and below 655 ft (200 m). The applicant evaluated the upper 655 ft (200 m) using geotechnical borehole and geophysical surveys. The profile below 655 ft (200 m) was derived from velocity profiles published by Gann-Phillips et al. (2024). Epistemic uncertainty was computed for the upper 655 ft (200 m) using the BE profile following EPRI (2013) and referenced profiles in Gann-Phillips et al. (2024) for below 655 ft (200 m). The NRC staff finds the applicant's approach to developing velocity profiles for the upper 655 ft (200 m)

and below 655 ft (200 m) technically reasonable based on the use of geotechnical borehole and geophysical survey data and literature review.

Evaluation of Site Kappa

The NRC staff reviewed the applicant's estimation of the high-frequency attenuation (site kappa) from four empirical equations. Campbell (2009) was weighted at 0.3 because of its applicability to CENA. Van Houtte et al. (2011) was weighted at 0.04, and Xue et al. (2020) was weighted at 0.16 due to both being based on Japanese data. Lastly, Guo and Chapman (2019) were weighted at 0.5 due to using data from the Coastal Plains and its model depending on sediment depth.

The NRC staff reviewed the applicant's site kappa estimation and determined the applicant's approach is technically reasonable because it uses established empirical models, accounts for differences in regional applicability among the models, and incorporates epistemic uncertainty through weighting. Therefore, the NRC staff finds the resulting kappa values appropriate and consistent with NRC-accepted methodologies (NRC 2021b).

Shear Modulus Reduction and Damping

The NRC staff reviewed the applicant's methodology for obtaining shear modulus and damping curve inputs for its site response analysis. The applicant derived modulus reduction and damping curves using the same properties and parameters for the last layer in the upper portion of the profile for the rest of the profile below 655 ft (200 m). Values of the bottom most layer in the upper 655 ft (200 m) were used for deriving material curves for layers below 655 ft (200 m), a standard approach for depths below geotechnical borehole data. The NRC staff also reviewed the applicant's approach to deriving the unit weight, plasticity index (PI), and overconsolidation ratio (OCR). PI was computed from boring data values for clay, silty sand, and poorly graded sand.

The NRC staff reviewed the applicant's material property curves and finds the methodology for obtaining shear modulus reduction and damping curves acceptable because it relies on site-specific data. For depths below 655 ft (200 m), where site-specific measurements are limited, the NRC staff finds it reasonable to characterize the deeper subsurface conditions using regional models derived from published studies. The NRC staff further finds that the logic tree approach appropriately accounts for epistemic uncertainty and is consistent with NRC recommendations (NRC 2021b).

Approach to Site Response Analysis

The NRC staff reviewed the applicant's methodology to capture uncertainty in site amplification using a site response logic tree. The first node of the logic tree addresses alternative methods of equivalent linear random vibration theory approach to site response analysis. The second node applies kappa correction factors to the kappa corrected equivalent linear results. The third node addresses epistemic uncertainty in velocity profiles between the 10^{th}, 50^{th}, and 90^{th} percentiles. The fourth node addresses soil nonlinear behavior. The fifth node addresses site kappa, which the applicant addresses in later subsections.

Randomization of Velocity Profiles and Material Curves

The NRC staff reviewed the applicant's process to randomize velocity profiles and material curves to account for aleatory variability due to spatial variability of materials. The applicant followed Steward et al. (2008) to calculate the number of sublayers and used layer thickness variation data obtained from the suspension log data. The applicant performed profile randomization using Toro (1995).

The NRC staff audited the approach to deriving site-specific shear wave (V_S) velocity profiles and performed a high-level examination of the data along with the geophysical and geotechnical testing locations, to assess uncertainty and variability. The NRC staff finds the derivation of the mean, lower, and upper range V_S profiles presented in the PSAR to be supported by the applicant's site profile data and is technically justified and acceptable. The NRC staff also reviewed the randomization of the velocity profiles and material curves and finds the applicant's analyses acceptable on a technical basis and consistent with NRC recommendations (2021b).

Broadened Conditional Mean Spectra

The NRC staff reviewed the applicant's approach to calculating the conditional mean spectra (CMS) targeted to reference rock hazard for AFE ranging from 3E-3 to 1E-9. The applicant applies adjustments to NGA-East median models for the LMGS site location.

The NRC staff reviewed the applicant's approach and use of broadened CMS to represent a more realistic characterization of the site's spectral shape. The NRC staff finds this approach acceptable on a technical basis because the applicant uses input values from the deaggregated site hazard analysis and made appropriate site-specific adjustments for regional effects.

Amplification Fitting

The NRC staff reviewed the applicant's approach to deriving the amplification functions obtained from each randomized profile, CMS input motion, and branch of the logic tree by dividing the output response spectra by the input response spectra and fit to the median SAFs. The applicant follows an approach recommended by the NRC (2021b) to account for epistemic uncertainty.

The NRC staff reviewed the applicant's approach to comparing the site-specific data derived SAF, and at AFE 1E-4 and 1E-5, the shape of the approximate and site-specific SAFs were similar but the amplitudes were different, with the approximate amplification exceeding the site-specific amplification across nearly the entire frequency spectrum. The applicant concludes that the approximate SAFs produce a conservative estimate of the GMRS.

The NRC staff reviewed the applicant's approach to deriving the SAF for LMGS. The NRC staff performed an independent analysis and produced comparable SAF values to those of the applicant. The NRC staff finds the applicant's SAFs acceptable on a technical basis because they are based on site-specific data and conform with the NRC-accepted methodology outlined in RG 1.208. The NRC staff audited the SAF data and performed a high-level analysis comparing the applicant's approximate and site-specific amplification functions, which show that the approximate SAFs conservatively bound the site-specific SAFs. The NRC staff finds the use of approximate amplification functions acceptable because it is validated by the site-specific analyses and sufficiently envelopes the site-specific SAFs.

2.5.2.5.2 **Conclusion**

The NRC staff finds that the methodology and analyses used to characterize seismic wave transmission of the site are well supported by the site-specific data, follow best practices as endorsed by NRC guidance and recommendations, and are comparable with NRC's independent verification analyses, indicating no significant differences of concern. The use of approximate site characterization is technically acceptable because the applicant substantiates these estimates with site-specific analyses and the approximate values sufficiently envelop the site-specific conditions.

2.5.2.6 Ground Motion Response Spectra

2.5.2.6.1 **Technical Evaluation**

The NRC staff reviewed the applicant's approach to develop site-specific mean and fractile seismic hazard curves. The applicant follows RG 1.208 to deaggregate the regional hazard results and follows INL (2022) to obtain deaggregation of the site-specific hazard. The applicant's analysis on contribution earthquakes implied a substantial contribution from distant earthquakes (>300 km).

The NRC staff reviewed the applicant's procedure for computing the horizontal and vertical GMRS. The applicant followed RG 1.208 to compute the GMRS at ground surface from the 10^{-4} and 10^{-5} uniform hazard response spectrum (UHRS). The applicant states that the PGA of the resulting GMRS is less than 0.1g but explains later in subsection 2.5.2.6.6 the modification made to the GMRS to meet regulations. The applicant references EPRI (2015b) to estimate the vertical GMRS using V_{S30} from the V_S profile and PGA.

The NRC staff finds the method that the applicant used to develop the UHRS technically acceptable because it is based on input rock hazard and SAFs. This SE documents the acceptability of the SAFs in preceding sections. The NRC staff also finds the fractile and mean hazard curves to be technically acceptable because their development is consistent with standard practice for PSHA.

The NRC staff finds the hazard deaggregation results technically reasonable and supported by the applicant's PSHA, which identifies EFFZ, Meers, and New Madrid sources as the highest contributors to the seismic hazard at LMGS.

The NRC staff reviewed the approach to developing the horizontal and vertical GMRS and determined that what was presented in the PSAR is acceptable on a technical basis because the applicant uses site-specific hazard inputs and NRC-endorsed standards.

<u>Safe Shutdown Earthquake Spectra</u>

The NRC staff reviewed the applicant's approach to developing the site's safe shutdown earthquake (SSE) used for the purpose of seismic design. The applicant did not derive the SSE as a single continuous spectrum but instead constructed it using different methodologies across distinct frequency ranges. Specifically, the applicant anchored the high-frequency portion to the RG 1.60, Revision 2 "Design Response Spectra for Seismic Design of Nuclear Power Plants" (ML13210A432), spectra at 0.1 g PGA, used a modeled ASCE 43-05 GMRS based on empirical

models and regional data in the intermediate frequency range, and used a modeled ASCE 43-05 GMRS based on site-specific data in the low-frequency range.

The NRC staff finds that, despite the departure from standard practice, the SSE is technically justified because it is based on site-specific input parameters, sufficiently envelopes site-specific values where regional data is used, and meets the requirement from Appendix S to 10 CFR Part 50 for a 0.1 g anchored PGA.

2.5.2.6.2 **Conclusion**

The NRC staff finds that the applicant's approach to developing the site's GMRS meets the requirements as stated in 10 CFR 100.23(d) to characterize the free-field ground motion response spectra at the free ground surface and is consistent with the guidance in RG 1.208. The staff further concludes that this GMRS provides an appropriate basis for deriving of the SSE used in the facility's seismic design in accordance with Appendix S to 10 CFR Part 50.

2.5.2.7 Conclusion

Based on NRC staff review of the information presented in PSAR section 2.5.2, the NRC staff concludes that the evaluation of vibratory ground motion hazard at the LMGS site satisfies the requirements of 10 CFR 100.23(c) because the applicant investigated and characterized the site's seismological characteristics in sufficient scope and detail to support an adequate evaluation of the site and its seismic hazards. The NRC staff further concludes that the evaluation satisfies the requirement of 10 CFR 100.23(d) because the applicant appropriately characterized the ground motion used to develop the site-specific GMRS, which provides the basis for the SSE. The analysis uses NRC-endorsed models with appropriate modifications and updates, is supported by site-specific subsurface characterization, and is consistent with current PSHA practices. The NRC staff further concludes that the SSE derived from the seismic hazard evaluation meets the requirements of section IV(a)(1)(i) of Appendix S to 10 CFR Part 50, in which the applicant is required to characterize the SSE with its corresponding response spectra and PGA anchored at 0.1 g. Therefore, the applicant has met the applicable regulatory requirements and has provided information that is consistent with applicable guidance for characterizing vibratory ground motion for the LMGS CP.

2.5.3 Surface Faulting

Section 2.5.3 of this SE addresses PSAR section 2.5.3. This SE is structured to provide the regulatory evaluation, the summary of the applicant's PSAR sections, and the NRC staff's technical evaluation.

2.5.3.1 Technical Evaluation

The NRC staff evaluated LMGS PSAR section 2.5.3 to determine whether the applicant adequately identified and evaluated potential sources of tectonic and nontectonic sources of surface deformation as required by 10 CFR 100.23(d). The applicant relied upon the VCS ESP application site safety analysis report (SSAR), Revision 1 (ML12131A060, ML12131A063, and ML12131A064); USGS-published maps; Texas' Bureau of Economic Geology-published maps; published literature; publicly available aerial photos and remote-sensing data; and its own field investigations to characterize deformation processes. The NRC staff finds that LME used

authoritative, published and unpublished sources of information and, therefore, the sources of information are adequate to characterize the potential sources of tectonic and nontectonic surface deformation.

Although PSAR section 2.5.3 is titled “Surface Faulting,” LME states that this section addresses multiple potential causes of surface deformation including bedrock faults, growth faults, karst collapse, salt tectonics, volcanism, and human-induced surface deformation. LME identifies and describes these features at the region (200-mile (322 km) radius around the site), vicinity (25-mile (40 km) radius around the site), area (5-mile (8 km) radius around the site) and location (0.6-mile (1 km) radius around the site). The NRC staff finds LME addressed an appropriate set of features and provided adequate detail across the various distances from the site. The NRC staff finds LME adequately described the site area to site region stratigraphy, geology and tectonic setting in PSAR section 2.5.1.

The LMGS proposed site is located roughly 20 mi (32 km) from the Exelon VCS and LME relied upon information in the VCS ESP application SSAR where the VCS and LMGS investigation areas overlap. The NRC staff finds the use of information from an existing application acceptable to characterize potential sources of surface deformation because the information was used to meet similar regulatory and guidance requirements and covered an area that partially overlapped the LMGS area.

The applicant adequately identifies the nearest basement-involved faults as the Balcones Fault zone and the Luling Fault Zone, both of which are outside the site vicinity, and describes the site region as experiencing “sparsely distributed instrumental seismicity, with most recent events likely linked to oil and gas extraction rather than natural seismicity.” The NRC staff finds this characterization is adequately supported by the information provided in PSAR sections 2.5.1 and 2.5.3.

The applicant reported that the LMGS site lies within the Frio Fault Zone and stated that growth faults in the region typically strike subparallel to the modern coastline. Accordingly, the applicant evaluated only lineaments exhibiting this orientation as potential growth faults. This evaluation included reviews of published maps, LiDAR, vertical aerial photographs, and field data. The NRC staff finds the applicant’s reasoning and investigation approach rational and, therefore, concluded that the evaluation of potential growth faults was adequate.

LME’s site field investigation yielded six samples from five site borings containing slickensides. LME provided more details on these slickensides during the audit, stating that additional slickensides are not found at similar depths nor in nearby borings. The applicant interprets these slickensides as isolated features and not indicators of growth faulting; therefore, LME concluded there is no evidence of Quaternary growth faults at the LMGS site. The NRC staff finds the applicant’s interpretation and conclusion to be adequately supported by the distribution of observed features as presented in the PSAR, especially sections 2.5.1.2.2 and 2.5.3.2.2.4.

The VCS ESP identified two inferred subsurface faults—GM-AD and GM-AE—within the LMGS site area. The applicant considered this information and evaluated additional nearby subsurface faults. LME limited the age of most recent displacement of these growth faults stratigraphically by noting that displacement associated with these growth faults is not mapped above the top of the Frio Formation (and therefore, displacement is pre-Quaternary) for the site area. The NRC

staff finds this conclusion reasonable because it is supported by the stratigraphic information presented for the site.

The applicant reported that InSAR and EigenSAR datasets indicate subsidence in the site area, but no differential offset or higher rates of subsidence at the LMGS site relative to the surrounding region. The applicant concluded that no subsidence hazard is present. Based on its review of the applicant's evaluation and supporting data, including data provided in response to audit questions that supported the conclusions in the PSAR, the NRC staff finds this conclusion acceptable.

LME discusses its compilation of an earthquake catalog of 343 earthquakes in the site region. After removing human-induced seismicity, the number of earthquakes decreased to 38 (including aftershocks). The applicant identified no natural earthquakes within 50 mi (80 km) of the site and no correlations with capable surface-faulting sources. The NRC staff determined that the applicant adequately evaluated human-induced seismicity and subsidence hazards and identified no resulting subsidence hazards at the LMGS site. The NRC staff also confirmed that no additional hazards related to salt tectonics, volcanism, or karst collapse were identified in PSAR section 2.5.3.

2.5.3.2 Conclusion

Based on results of the review and evaluation of the detailed information presented in PSAR section 2.5.3 and discussed above, the NRC staff finds that the applicant investigated, evaluated, and established the geologic characteristics of the LMGS site region, site vicinity, site area, and site location in accordance with 10 CFR 50.34(a)(1)(ii)(D), 10 CFR Part 50 Appendix A, Section I, criterion 2, 10 CFR 100.21(d), 10 CFR 100.23(c), and 10 CFR 100.23(d), and in conformance with guidance in NUREG-800 and RG 1.208.

2.5.4 Evaluation of Subsurface Stability

2.5.4.1 Technical Evaluation

Site Characterization

LME described the subsurface investigation program at the proposed site in PSAR section 2.5.4.2 and the laboratory tests conducted on samples collected at the site in PSAR section 2.5.4.2.2. In addition, PSAR section 2.5.4.2.1 described field investigations at the proposed site.

PSAR section 2.5.4.2.1 states that LME drilled 57 primary soil borings at the proposed site reaching depths ranging from 100 ft to 669 ft (30.5 m to 204 m) below the ground surface (bgs). PSAR figure 2.5.4-1 shows the location of these borings at the proposed site. Supplemental borings were drilled to obtain samples or conduct geophysical testing. Standard penetration test (SPT) soundings were done in these borings. Additionally, LME conducted 27 cone penetration test (CPT) soundings extending to the refusal depths ranging from 73.5 ft to 953.7 ft (22.4 m to 290.7 m) bgs. In addition, P-S logging and other geophysical testing were conducted in 4 borings and 8 surficial lines were established to conduct the spectral analysis of surface waves (SASW) to determine the shear wave velocity. PSAR table 2.5.4-1 lists the location and depth of each boring. Cone penetration testing with pore pressure measurements (CPTu) was conducted

at 27 locations. In addition, seismic cone penetration testing (SCPTu) was conducted at 5 locations of the proposed site.

The NRC staff finds that LME drilled a sufficient number of borings within the proposed site following the guidance in RG 1.132 “Geologic and Geotechnical Site Characterization Investigations for Nuclear Power Plants” (ML21298A054). Samples were collected from these borings to measure different properties in the laboratory. SPT soundings were conducted in these boreholes as the holes were advanced. Additionally, CPT soundings were conducted in an additional 27 borings. These measurements are discussed in detail below.

Static Properties of Subsurface Strata

LME conducted laboratory tests to determine different soil parameters using split spoon, bulk, and tube samples of soil obtained at the proposed site. The NRC staff finds that the laboratory tests were conducted following industry standard methods; for example, American Association of State Highway and Transportation Officials standards, American Society for Testing and Materials (ASTM) standards, or other appropriate standards, as listed in PSAR section 2.5.4.2.2. PSAR section 2.5.4.2.2 also lists the laboratory tests performed. The NRC staff finds that LME has used appropriate standards to conduct the laboratory tests to determine the soil properties using samples collected from the proposed site. The tests measured a comprehensive suite of soil properties, as listed in PSAR section 2.5.4.2.2.

The proposed site is underlain by the Beaumont Formation, which consists of fluvial-deltaic sediments deposited during the Pleistocene period. The sediment deposits are approximately 30,000 ft (9144 m) thick and overlie the continental crust. Based on the geotechnical properties measured of samples collected, LME identified 13 distinct soil strata beneath the proposed site up to a depth of 660 ft (201 m), as described in PSAR section 2.5.4.2.3. PSAR table 2.5.4-1 lists the generalized depth and materials representing each stratum. PSAR table 2.5.4-1 also presents the SPT blow counts (N_{60}) corrected for hammer energy for each stratum. The strata alternate between clay and sand with increasing depth from surface. Stratum I, just below the surface, is clay. The next stratum II is sand and underlies stratum I. The strata were identified based on their physical and engineering characteristics including SPT blow count. PSAR tables 2.5.4-6 through 2.5.4-16 list the soil type identified in each stratum along with measured Atterberg limits and other material properties. Materials in each stratum were classified following the universal soil classification system (USCS) and used to identify each stratum, as listed in PSAR table 2.5.4-2 through 2.5.4-16.

PSAR tables 2.5.4-6 through 2.5.4-16 also list the results obtained in laboratory tests using samples collected from the 13 soil strata identified at the proposed site. Results shown in these tables show variation of the material properties due to change in the deposition of the sediments. LME used the average value of a property measured in the samples obtained from each stratum.

The NRC staff finds that LME has characterized the subsurface and measured properties of clay and sand stratum appropriately to a depth sufficient for the proposed facility compared to the soil embedment depth of the drilled piers to be used at the proposed facility. Therefore, the NRC staff also finds that the measured properties of clay and sand strata can be used in designing the proposed facility.

Excavation and Backfill

PSAR section 2.5.4.3 states that the only SR structure at the proposed facility is the reactor support structure contained within the reactor shield building. The reactor shield building along with the reactor support structure will be on pier caps (mat foundation). Drilled piers will support the mat foundation. Skin friction of these piers provides the major portion of the axial compression and tension capacity for the foundation. End bearing provides the rest of the capacity. These drilled piers extend from stratum I to stratum VI and will bear on stratum VI, as described in PSAR section 2.5.4.3.

As stated in PSAR section 2.5.4.5, the SR structures will be constructed above grade. As stated in PSAR section 2.5.4.6, the site will be elevated from a nominal grade of 27.0 ft to 31.5 ft (8.2 m to 9.6 m) NAVD 88 using structural fill. As the SR structures will not be embedded, only SR backfill will be adjacent to the pier caps. Excavations at the proposed site will generally be limited to the foundation level (e.g., pier caps). The slope of these excavations will be in accordance with the US Department of Labor–Occupational Health and Safety Administration regulations. Groundwater in the excavations will be controlled using sumps and pumps and will be disposed of in a proper manner using the Best Management Practice. The dewatering system will have sufficient capacity to conduct the excavation in dry conditions, as stated in PSAR section 2.5.4.6. Based on the above discussion, the NRC staff finds that construction of the shallow excavations will follow appropriate guidance.

PSAR table 2.5.4-19 lists the properties acceptable for granular structural fill to be used in construction at the proposed facility. PSAR section 2.5.4.5 states that the structural fill would be compacted to a minimum of 95 percent of its maximum dry density, measured using standard Proctor method within ±3 percent of its optimum moisture content, measured following ASTM Standard ASTM D1557, “Standard Test Methods for Laboratory Compaction Characteristics of Soil Using Modified Effort.” In PSAR section 2.5.4.5, LME commits to conducting periodic density testing of the compacted fill as it is placed.

PSAR section 2.5.4.5 states that the fill materials will be obtained from on-site borrow pits in stratum I, highly plastic CH materials. PSAR section 2.5.4.2.3.1 shows that stratum I is approximately 62.5 ft (19.1 m) thick. Consequently, the NRC staff finds the fill material is expected to be available in sufficient quantities excluding the unusable material lenses in stratum I.

Clay in stratum I may require stabilization. LME proposes to mix lime or cement with the soil to counteract the plasticity and soil shrinkage/expansion characteristics. LME will develop a construction materials testing program to verify whether soil improvement is needed and soil improvement through lime or cement can achieve the required properties. LME will identify location of the site borrow pit during final design stage, as stated in PSAR section 2.5.4.12. LME anticipates the testing will include measurements of the index properties, sulfate concentration, and improvement of soil properties with lime and/or Portland Cement. LME stated in PSAR section 2.5.4.12 that the methods adopted to test the backfill materials will be based on widely accepted industry standards, such as Texas Department of Transportation Test Procedure for Cement Treated Materials (Tex-120-E) and Test Procedure for Lime Treated Materials Tex-121-E. The NRC staff finds these test procedures are widely used in the industry to characterize soil after being treated with cement or lime to improve its properties and, therefore, is acceptable.

Based on above discussion, the NRC staff finds that LME proposes to use structural fill for foundation support and backfill the gap between the walls (excavation and structural walls). The fill is expected to be from stratum I and will be compacted following appropriate industry standards. Additionally, the NRC staff finds that LME proposes to develop a testing program to measure the characteristics of fill using appropriate industry standards. Therefore, the NRC staff finds that LME has described its plans in sufficient detail to excavate the site for construction of the SR structures. In addition, LME has identified the characteristics of the fill materials to be used in PSAR table 2.5.4-19 and its source (Stratum I). Additionally, LME will establish a testing program to test the fill materials for suitability and identify any modifications that need to be made to the fill materials.

Foundation Design and Bearing Capacity

PSAR sections 6.4.1, 6.4.2, and 7.3.1 describe the foundations of the SR and NSRST structures to be built at the proposed facility. PSAR section 2.5.4.10 discusses the foundation design of the SR structures. As stated in PSAR section 2.5.4.3, the reactor shield building contains the reactor support structure, the only SR structure in the proposed facility. However, there are NSRST structures that contain SR equipment. The bearing capacity for these structures will need to be confirmed in a similar manner to support the design analysis described in PSAR section 3.7 and evaluated in section 3.7 of this SE.

PSAR section 2.5.4.3 also states that the SR structures at the proposed facility will be resting on pier cap, a thick mat foundation, supported by drilled piers which will extend from strata I through VI and will bear on stratum VI. LME has not yet selected the construction methodologies to construct the drilled piers; however, LME plans to construct the piers under slurry to protect the end bearing zone in sand stratum. The NRC staff finds the level of detail provided on the foundation design and bearing capacity is acceptable for issuance of a CP. At the OL application, the NRC staff would review the review the excavation techniques in accordance with 10 CFR 50.34(b).

As stated in PSAR section 2.5.4.7.1.1, the grade at the proposed site will be raised 5 ft (1.5 m) by placing fill materials and compacting them. Although raising the grade of the site increases skin friction of the pier shaft and, thereby, its axial capacity, LME neglected the contribution of the raised grade to skin friction in estimating the axial capacity. The NRC staff finds that this is a conservative assumption and is therefore acceptable for estimating the axial capacity of the pier shafts.

PSAR section 2.5.4.10.1 states that the drilled piers will have a permanent steel casing at the upper section, filled entirely with concrete, to provide strength against structural bending moment. As the permanent casing will extend through the cohesionless sand layers, a casing reduction factor of 50 percent is assumed for the skin friction value for the cased section of the drilled piers. PSAR section 2.5.4.10.1 also shows that LME estimated the external diameter of the cased section accounting for corrosion. As side resistance provided by a pier will be reduced due to the permanent casing, LME considered only half of skin friction of uncased cast-in-place concrete piers provided by the cased section.

LME used the methodologies given in USDOT, Federal Highway Administration (FHWA), “Drilled Shafts: Construction Procedures and Design Methods,” FHWA-NHI 18-024, 2018. The NRC staff finds that the methodologies in FHWA (2018) are widely used in the industry and,

therefore, acceptable. The NRC staff concludes that LME has used appropriate methodologies to design the drilled pier foundation system for the SR structures at the proposed facility. In addition, LME assumed a factor of safety of 2.0 for both axial compression and tension capacities of the drilled piers to estimate the allowable capacity, as given in PSAR section 2.5.4.10.1.

LME used the stratigraphy and material properties described in PSAR section 2.5.4.2.1 to assess the bearing capacity and settlement of the proposed foundation system. Strata-specific properties are described in PSAR tables 2.5.4-1 through 2.5.4-7 and sections 2.5.4.2.1.1 through 2.5.4.2.1.6. The ultimate drilled pier capacity was determined by summing contribution from each pier segment in each stratum considering all assumptions discussed above.

As stated in PSAR section 2.5.4.2.1, the allowable axial compression capacity was determined to be 5,100 kips (22,690 kilonewtons (kN)) assuming a factor of safety (FS) of 2.0. Similarly, the allowable axial tension capacity is 3,400 kips (15,120 kN) with an FS of 2.0. PSAR table 2.5.4-20 presents the calculated maximum axial tension, and compression loads under seismic and static conditions. This table also lists the allowable lateral loads under both static and seismic conditions. As discussed in PSAR section 2.5.4.10.1, the designed foundation system has significantly higher capacity to withstand demand (loads) posed by the structures they support. During construction, LME commits to implement a best practice load test program to validate the soil resistance values used in the analysis.

PSAR section 2.5.4.10.2 describes the settlement assessment of the mat foundation on drilled piers. Following FHWA (2018), LME assumed the drilled piers act as a group. The total settlement of the group of drilled piers consists of settlement due to soil compression and settlement due to elastic shortening of the drilled piers due to applied load. Following FHWA (2018), LME estimated the group settlement from soil compression, assuming an equivalent footing located at one-third of the embedment depth above pier toe elevation. This equivalent footing has the plan area defined by the pier group length multiplied by the pier group width. The load from the structures is spread over this area and the pressure applied by the load is transferred to the underlying soil through the equivalent footing. The stress distribution within the underlying soil is estimated assuming the bearing pressure reduces with depth as the area increases. LME used the 1 : 2 (vertical : horizontal) method to estimate the area increase and pressure decrease with depth. The NRC staff finds that this equivalent footing concept was originally proposed by Terzaghi and Peck (1967) and is widely used to analyze a group of drilled piers (also, pile groups) and is therefore acceptable.

The calculated stress at each stratum is used to determine its settlement due to application of the loads imposed by the structure at the surface. LME used the consolidation properties of the cohesive strata (clay) measured using one-dimensional consolidation test, as discussed in PSAR section 2.5.4.2.3, to estimate the settlement of the clay strata at the proposed site. Settlement of the sand layers were estimated using the elastic compression theory. The NRC staff notes that magnitude of the stress increase imposed on a stratum by the load of the structures decreases with depth. Typically, a cut-off is assumed when the stress increase from loads imposed by the structure is less than 20% of the original stress from stratum self-weight. Therefore, LME has considered several strata below stratum VI to estimate the settlement. The NRC staff notes that the drilled pier system will bear upon stratum VI. The total settlement of the foundation is the sum of settlement estimated for each stratum.

As discussed in PSAR section 2.5.4.10.2, the SR RSS and the reactor shield building will provide the load applied on the foundation. PSAR section 2.5.4.10.2 states that the axial load per drilled pier was 2,500 kips (11,120 kN). For settlement estimation, this load is assumed to be uniformly distributed over the entire foundation, estimated to be 7.0 kips per square foot (335 kN/m^2). LME estimated the settlement from soil compression to be 3.6 in. (9.1 cm). In addition, drilled piers have shortened by 0.2 in. (0.51 cm) elastically. Consequently, the total settlement estimated is 3.8 in. (9.7 cm).

The NRC staff finds that LME has used appropriate methodologies to estimate both the bearing capacity and associated settlements. LME has used an appropriate strata sequence developed from site-specific investigation in this assessment. The material properties of these strata were used to estimate the bearing capacity of the proposed drilled piers-mat foundation and the settlement the foundation will sustain as the result of constructing the SR structures at the proposed site. Both methods for estimating the bearing capacity and the settlement are widely used in the industry and, consequently, appropriate to use. Results obtained show that the proposed foundation will be able to sustain the load and undergo tolerable settlement. In addition, the NRC staff notes that LME commits to implement a load test program to validate the soil resistance values used in the analysis.

Liquefaction Assessment

In PSAR section 2.5.4.8, LME described the potential for liquefaction at the proposed site. The NRC staff finds that LME evaluated the potential for liquefaction following the guidance given in RG 1.198 “Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites” (ML033280143) using SPT, CPT, and shear wave velocity (Vs) data with empirical methods proposed in the literature. The controlling earthquake with magnitude M6.5 and PGA of 0.1 g defined the seismic loads in the subsurface. The groundwater table is taken at a depth of 2 ft (0.6 m) below the ground surface (bgs), as stated in PSAR section 2.5.4.8. Properties of the soil layers are described in PSAR sections 2.5.4.1, 2.5.4.2, and 2.5.4.4.

As the water table is only 2 ft (0.6 m) bgs, the soils at the proposed site are considered saturated. As noted in RG 1.198, earthquake-induced liquefaction is common in fluvial and alluvial soil deposits, eolian sands and silts, and beach sands. The greatest potential for liquefaction occurs in recent Holocene deposits and uncompacted fills (National Academies of Science, Engineering, and Medicine, 2016; Seed, et. al., 2003). As the proposed site consists of fluvial-deltaic sediments belonging to the Quaternary Beaumont Formation deposited during the Pleistocene epoch, LME concluded that upper 655 ft (200 m) of soils at the proposed site are potentially susceptible to liquefaction. As the soils below 655 ft (200 m) are geologically old, the deeper sediments will be generally non-liquefiable. Following RG 1.198, liquefaction can be triggered if the estimated FS, defined in RG 1.198, is less than or equal to 1.1.

SPT Recordings

As stated in PSAR section 2.5.4.8.2, a total of 57 primary soil borings were conducted at the proposed site. A total of 2,849 SPT recordings were made at these 57 borings. The deepest SPT recordings were made at depths of approximately 655 ft (200 m). The recorded blow counts were corrected, as recommended in Youd et al. (2001).

LME used the method proposed by Bray and Sancio (2006) to screen out all CL (lean clay) samples, as per USCS classification, as they are susceptible (clay-like) to liquefaction. Based on this susceptibility assessment, only 833 out of 2,849 SPT samples taken at all depths were considered as potentially susceptible to liquefaction. As liquefaction occurs at shallow depth, LME used a depth cut-off of 80 ft (24 m) to identify 153 SPT samples out of 1,322 samples within 80 ft (24 m) from the surface to be potentially susceptible to liquefaction. The NRC staff finds the cut-off depth of 80 ft (24 m) to be conservative using the results of Idriss and Boulanger (2008), Boulanger and Idriss (2014), and Cetin et al. (2009).

LME used three methods to estimate the FS against liquefaction: (1) Youd et al. (2001), (2) Boulanger and Idriss (2014), and (3) Cetin et al. (2016). An increase in vertical stress from raising the current grade by 5 ft (1.5 m) of fill was included in the assessment, as stated in PSAR section 2.5.4.5. The FS values against liquefaction, calculated using Youd et al. (2001) and Boulanger and Idriss (2014), are always larger than 1.1; however, three SPT samples collected at depths between 65 ft to 70 ft (20 m to 21 m) from borings X15, X22, and X25 show FS less than 1.1 using the method in Cetin et al. (2016), as shown in PSAR table 2.5.4-39. LME argued that it is unlikely that liquefaction would be triggered for these SPT samples as they are isolated points in the subsurface. Using RG 1.198, LME identified that the potential for liquefaction at these locations is limited in spatial extent. The NRC staff finds that following RG 1.198 is acceptable for determining the potential for liquefaction at these locations.

CPT Recordings

As discussed before, LME conducted 27 CPT tests at the subsurface of the proposed site from depths ranging from 73.5 ft to 153.7 ft (22.4 m to 46.8 m). The deepest CPT tests were at depths approximately 150 ft (45.7 m) from the surface. The field-measured CPT data were corrected following Boulanger and Idriss (2014) to generate q_{c1N} values (measured cone tip resistance values normalized for overburden pressure).

The NRC staff finds that in PSAR section 2.5.4.8.3, LME used the methodology suggested by Boulanger and Idriss (2014, 2015) to identify susceptibility to liquefaction using CPT measurements and concludes that LME has used an acceptable and widely used methodology. This methodology calculates the cyclic resistance ratio, as defined in RG 1.198, using normalized parameters. As discussed before, the vertical stress at any location of the proposed site included stress from a 5-ft (1.5 m) thick fill to be used to raise the elevation of the final site surface, as stated in PSAR section 2.5.4.5. As stated in PSAR section 2.5.4.8.3, all 27 CPT tests have a FS against liquefaction greater than 1.1. Therefore, the NRC staff agrees that these locations will not liquefy during the design-basis earthquake.

Vs Recordings

As discussed before, LME conducted geophysical investigation of the proposed site, as stated in PSAR section 2.5.4.4. Four suspension P-S logging velocity measurements were conducted in Borings X101A, X102A, X103A, and X94A. In addition, five seismic CPT measurements were also conducted. The Vs data were corrected for the vertical effective overburden stress, as given in Kayen et al. (2013), and the methodology proposed by Kayen et al. (2013) was used to identify susceptibility to liquefaction. The NRC staff finds that LME has used an acceptable method to normalize the Vs data measured at different depths. As discussed before, the vertical stress is increased by the assumed fill of 5 ft (1.5 m) to raise the current grade. LME concluded

that all points where Vs was measured at the proposed site are not susceptible to liquefaction as the calculated FS was greater than 1.1, except at two points in boring X9A. However, the samples were classified as clay, either CH or CL. As CH or CL soils do not liquefy, as stated in RG 1.198, all locations in the proposed site are found not to be susceptible to liquefaction.

Sediments Deeper than 655 ft (200 m)

As the sediment cover over the base rock beneath the proposed site is extremely thick, approximately 30,000 ft (9,144 m) as described in PSAR section 2.5.1.2.2, LME also assessed the susceptibility of the sediments at deeper depths to liquefaction using knowledge of site geology and Vs measurements. As discussed, measurements of SPT and CPT soundings were limited to the upper 655 ft (200 m). However, the deepest Vs measurements were made at a depth of approximately 900 ft (274 m) using the SASW array. Consequently, LME could assess the susceptibility up to approximately 900 ft (274 m) using geological information coupled with Vs measured data.

As stated in PSAR section 2.5.1.2.3, soils in the upper 5,200 ft (1,585 m) at the proposed site are from Miocene epoch (23 to 5 million years ago) and younger. Sediments from Holocene Guadalupe River fluvial-deltaic system are near the surface of the proposed site overlying Pleistocene (2.58 million to 11,700 years ago) Beaumont Formation. Resistance to liquefaction increases markedly with the geological age of the soil, as stated in Youd et al. (2001). Therefore, the NRC staff finds the potential for liquefaction in pre-Pleistocene soils is rare. Additionally, the NRC staff finds Vs at the proposed site below 655 ft (200 m) to a depth of approximately 5,000 ft (1,524 m) to be 2,000 to 4,000 ft/s (610 to 1219 m/s), as discussed in PSAR section 2.5.4.7.1.2. It is expected that the degree of compaction and the strength of soil increase with depth. Therefore, the NRC staff finds that liquefaction below a depth of 655 ft (200 m) at the proposed site is not possible.

Summary/Conclusion of Liquefaction Review

The NRC staff finds that LME assessed the potential for liquefaction of the subsurface at the proposed site using measurements of SPT, CPT, and Vs values with depth for a design-basis seismic PGA of 0.1g. The methodologies used with each measurement type are widely used in the industry and, therefore, are acceptable. Results show that the subsurface of the proposed site is not liquefiable. Only 5 samples showed any potential for liquefaction, and they were assessed as non-liquefiable after further assessment using other methods. In addition, samples surrounding these locations do not show any susceptibility to liquefaction. Therefore, the NRC staff finds that even if a sample location below ground liquefies, the effects would be extremely localized and small and the effects manifested on the surface would be tolerable to the structures above these locations.

In addition, in PSAR section 2.5.4.8 LME reported the estimated FS to be above 1.1 to indicate no potential for liquefaction. Section 3.2 of RG 1.198 states that an FS above 1.1 but below 1.4 requires additional evaluation for progressive failure. During audit, LME confirmed that most of the soil elements analyzed have an FS greater than 1.4. Only a small fraction (less than 20) have an FS between 1.1 and 1.4. These elements are dispersed throughout the site, both horizontally and vertically. Therefore, the NRC staff concludes that even if these samples fail in cyclic stress, the failure will be localized and isolated, and therefore will not impact the structures above these locations. The NRC staff also notes that soils at the proposed site are from the Pleistocene epoch or older. As discussed in Youd et al. (2001), soils from Pleistocene

epoch are more resistant to liquefaction. Therefore, the NRC staff finds that the soils at the proposed site are not likely to liquefy in a seismic event. Therefore, the zone of liquefaction, even if it formed in isolated points at the subsurface, is not expected to grow or impact the structures at these locations.

2.5.4.2 Conclusion

The NRC staff finds that LME has adequately characterized the subsurface of the proposed site using guidance in RG 1.132 and 1.38 and other applicable industry standards and used that information to show that the proposed site does not pose liquefaction hazard to the proposed facility. Therefore, the NRC staff finds that the requirements of 10 CFR 100.21(d) have been satisfied. LME has also shown that the proposed foundation design will have adequate bearing capacity to sustain the loads imposed by the structures without undergoing excessive settlement and, therefore, has satisfied the requirements of 10 CFR 100.23(d)(4) with respect to soil and rock stability and the potential for liquefaction.

2.5.5 Stability of Slopes

2.5.5.1 Technical Evaluation

The proposed site is relatively flat with no significant natural slopes. The NRC staff conducted a site visit and confirmed the description of the site by observation. However, LME is proposing to develop several sloped areas as part of the construction:

1. SR fill materials will be used to elevate the existing grade of the NI area by a nominal 5 ft (1.5 m) to a final grade of 31.5 ft (9.60 m). As stated in PSAR section 2.5.5, the slope will be constructed at a grade of 4 horizonal to 1 vertical (4H : 1V) (approximately 14° slope) at least 100 ft (30.5 m) from the SR structures. This is a shallow slope.

2. Surface water ditches and distension basins with less than 15 ft (4.6 m) depth may be graded with 4H:1V slopes, as stated in PSAR section 2.5.5. In addition, these ditches and distension basins will be at least 100 ft (30.5 m) away from SR structures.

As stated in PSAR section 2.5.5.2, the Spencer method (USACE, 2003) was used to assess the stability of these slopes at the end of construction (Case 1) and long-term conditions (Case 2). A typical surcharge load of 500 psf was assumed in Case 3. A downward vertical ground acceleration with horizontal acceleration was considered in Case 4. Case 5 considered an upward vertical acceleration with horizontal ground acceleration in the analysis. The NRC staff finds that the Spencer method is commonly used for slope stability assessment and is acceptable to the NRC staff. Therefore, the NRC staff conclude that LME has used an appropriate method to assess the stability of the slopes in the proposed facility.

Following US Navy Facilities Command NAVFAC (1986), the calculated static FS should be at least 1.3 (Case 1) and 1.5 in long term (Case 2). The FS under surcharge load (Case 3) must be at least 1.3 and during seismic events (Cases 4 and 5) must be at least 1.15. Results given in PSAR section 2.5.5.2 show that the estimated FS for Cases 1 through 5 is more than the minimum specified.

In addition, PSAR section 2.5.5.5 states that as the proposed site is relatively flat, any possibilities of natural topographic slope stability from human-induced surface deformation would be limited and is not expected to impact the safety of the proposed facility. The NRC staff agrees with the applicant's assessment and finds it reasonable.

2.5.5.2 Conclusion

In summary, the NRC staff finds that the proposed site is relatively flat. LME has shown through slope stability analyses using an appropriate method that construction-related slopes will be stable even during an earthquake as the design has significant FS. Consequently, the NRC staff finds that LME has satisfied the requirements of 10 CFR 100.23(d)(4) with respect to natural and artificial slope stability.

3 METHODOLOGIES, ANALYSES, AND LICENSING BASIS EVENTS

This chapter of the safety evaluation (SE) summarizes the U.S. Nuclear Regulatory Commission (NRC or Commission) staff's review and evaluation of the Long Mott Generating Station (LMGS) preliminary safety analysis report (PSAR) chapter 3, "Methodologies, Analyses, and Licensing Basis Events."

3.1 Methodologies and Analyses

3.1.1 Probabilistic Risk Assessment

PSAR section 3.1.1 summarizes the scope, methodology, and technical adequacy of the probabilistic risk assessment (PRA). The PSAR section discusses how the advanced non-light-water reactor (non-LWR) PRA standard was applied to LMGS and includes a summary of the results from the PRA self-assessment. The PRA is used as the primary tool for implementing the risk-informed and performance-based licensing modernization project (LMP) methodology described in Nuclear Energy Institute (NEI) 18-04, Revision 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," (ML19241A472), as endorsed with clarifications in Regulatory Guide (RG) 1.233, Revision 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.". RG 1.233 includes guidance for identifying licensing basis events (LBEs), determining the classification of structures, systems, and components (SSCs) and their special treatments, and evaluating defense-in-depth (DID) adequacy.

The LMGS PRA provides an integrated assessment of overall plant risk against the following risk evaluation criteria described in NEI 18-04, section 3.2.2, step 7b, and addressed in PSAR section 4.1:

- The total mean frequency of events exceeding a site boundary dose of 100 millirem (mrem) (1 mSv) from all LBEs should not exceed one/plant-year;
- The average individual risk of an early fatality within one mile of the exclusion area boundary (EAB) from all LBEs should not exceed $5x10^{-7}$/plant-year; and
- The average individual risk of latent cancer fatalities within ten miles of the EAB from all LBEs should not exceed $2x10^{-6}$/plant-year.

The applicable regulatory requirements at the construction permit (CP) application stage related to the use of PRA to inform the licensing basis of a non-LWR design and the risk to public health and safety from plant operation are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a); and
- 10 CFR 50.35, "Issuance of Construction Permits."

The relevant guidance documents related to the use of PRA for this CP application includes:

- RG 1.233, Revision 0, which endorses the guidance in NEI 18-04, Revision 1, with clarifications;

- RG 1.247, “Acceptability of Probabilistic Risk Assessment Results for Non-Light-Water Reactor Risk-Informed Activities,” For Trial Use (ML21235A008), which endorses the non-LWR PRA standard ASME/ANS RA-S-1.4-2021; and

- RG 1.253, Revision 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 NEI 21-07, Revision 1, “Technology Inclusive Guidance for Non-Light Water Reactor Safety Analysis Report: For Applicants Utilizing NEI 18-04 Methodology,” (ML22060A190), with clarifications and additions.

NEI 18-04 describes an expanded role for PRA for non-LWRs beyond current Commission policy for potential applications under 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities.” Following NEI 18-04, the applicant uses the PRA to develop or confirm the ESs, safety functions, key SSCs, and adequacy of DID. Because the PRA plays an important role in the plant design process, the NRC staff audited how LME developed and used the PRA to develop the preliminary LMGS licensing basis following the LMP approach and applied specific risk insights to the LMGS design. The NRC staff followed the guidance in appendix A of RG 1.253 to audit the LMGS CP stage PRA within the context of the LMP framework. Appendix A provides guidance on the expected scope, level of detail, and quality of a PRA suitable for the CP stage, recognizing that the plant design is still evolving and that the PRA model is inherently limited in maturity. The NRC staff applied RG 1.247, which endorses, with clarifications, the American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) RA-S-1.4-2021 non-LWR PRA standard, to evaluate the technical adequacy and completeness of each PRA element performed. The NRC staff’s review of the PSAR therefore focused on assessing whether the PRA provides a technically sound foundation to support LMP-related activities, specifically, the identification and evaluation of LBEs, classification of SSCs according to safety significance, and the establishment of adequate DID. At the CP stage, the NRC staff determines whether the preliminary PRA follows RG 1.233 and appendix A of RG 1.253, thereby providing a provisional basis for assessing risk acceptance criteria (e.g., quantitative health objectives (QHOs), frequency-based thresholds for LBEs), and to inform early-stage design and licensing decisions through systematic risk insights, given that uncertainties related to parameter, model, and completeness have not yet been fully characterized or assessed. The NRC staff’s audit focused on whether the PRA in use at the CP stage provides a reasonable technical basis for ongoing design development and establishes a foundation for future updates and expanded scope at the operating license (OL) application stage.

The PRA insights and results described in PSAR section 3.1 include inputs to multiple PSAR chapters where risk information is used as an input to the LMGS design and safety analysis. In PSAR section 4.1, LME incorporates PRA results into the overall assessment of plant-level risk and in PSAR section 4.2, LME describes its assessment of the adequacy of DID. PSAR section 5.1 describes LME’s application of PRA insights to the classification of SSCs. Similarly, in PSAR section 5.2, LME identifies the safety-significant PRA safety functions (PSFs), including safety-related (SR), non-safety-related SSCs with special treatment (NSRST), and

non-safety-related with no special treatment (NST) functions derived from PRA results. In PSAR sections 6.2 and 7.1, LME describes the use of PRA data to establish reliability and capability targets. In PSAR sections 6.4 and 7.3, LME applies PRA-derived safety functions to define functional requirements and design criteria, and PSAR section 11.4 describes the use of PRA insights to support analyses related to plume exposure pathway emergency planning zone size. As such, the PRA described in PSAR section 3.1 is instrumental in supporting development of the LMGS LMP licensing basis, ensuring that the design, classification, and operational evaluations are systematically informed by quantitative risk insights and aligned with the overall risk-informed licensing approach.

3.1.1.1 Technical Evaluation

PSAR section 3.1 summarizes the LMGS CP stage PRA and introduces how the PRA results and insights are linked to safety analysis and design evaluation throughout the PSAR. Specifically, this section presents the PRA overview, key modeling assumptions, and a summary of PRA results essential to the licensing basis under the LMP framework. The information in PSAR section 3.1, supports the risk-informed identification of LBEs and provides the quantitative basis for determining safety-significant functions, establishing DID, and defining reliability and capability targets for important SSCs. The NRC staff considered the applicable regulations and guidance identified above to determine whether the LMGS PRA was developed using generally accepted methods and reflects an appropriate level of technical rigor to support its intended use at the CP stage. Specifically, the NRC staff evaluated PSAR section 3.1 against RG 1.233, RG 1.253 sections A.3 and A.5, and RG 1.247. These guidance documents outline the acceptable practices for PRA development and application within the framework of the LMP and provide expectations for the PRA scope and level of detail in support of a CP application. In particular, RG 1.253 sections A.3 and A.5 indicate that the PRA description submitted with a CP application be a high-level overview or summary that addresses PRA quality, scope, intended applications, and overall acceptability. The applicant is expected to demonstrate that the PRA was developed in a manner commensurate with the maturity of the design, and that the PSAR includes sufficient information to support the CP-level safety determinations.

The NRC staff assessed the acceptability of the LMGS PRA by evaluating the PRA scope, level of detail, PRA elements, and plant representation and configuration control in an integrated manner. The NRC staff's review of the PRA scope includes the risk metrics used, the radiological source terms considered, the internal event causes (hazard groups) analyzed[1]; and the plant operating states (POSs) evaluated.

The NRC staff's review of the level of detail includes the extent of modeling necessary to capture plant behavior, system interactions, and interdependencies with sufficient resolution to support intended applications.

The NRC staff's review of the PRA elements includes the technical analyses that support the development and quantification of the PRA model, consistent with its intended use at the CP stage.

[1] As described in appendix A of RG 1.253, an applicant for a CP may disposition certain hazards by crediting design basis hazard levels (DBHLs) in lieu of explicitly modeling these hazards in the PRA or accounting for them through a risk-informed supplementary evaluation.

The NRC staff's review of the plant representation and configuration control includes the degree to which the PRA accurately reflects the current design and the mechanisms in place to manage configuration changes as the design evolves.

Consistent with RG 1.253, the PSAR describes the intended use of the LMGS PRA during the design and construction phases:

- PSAR chapter 3, "Methodologies, Analyses, and Licensing Basis Events," provides a high-level overview of the PRA model and presents the set of LBEs, which are derived from and informed by PRA event sequence (ES) families.

- PSAR chapter 4, "Integrated Evaluations," presents the integrated risk results and compares them to the three cumulative plant performance criteria outlined in NEI 18-04.

- PSAR chapter 5, "Required Safety Functions, PRA Safety Functions, and SSC Classifications," identifies the preliminary set of PSFs, including those fulfilled by both SR and NSRST SSCs. The chapter also documents risk-significant SSCs credited for meeting frequency-consequence (F-C) targets and cumulative risk criteria.

- PSAR chapter 6, "Safety-related SSC Criteria and Capabilities," and PSAR chapter 7, "Non-Safety-Related with Special Treatment Structures, Systems, and Components Criteria and Capabilities," discuss the reliability and capability targets for SR and NSRST SSCs and describe how PRA results and insights informed the development of these targets and their associated special treatments.

The NRC staff's evaluations and conclusions regarding the PRA-related content presented in other sections of the PSAR are documented in the corresponding sections of the NRC staff's SE. This section focuses specifically on the scope, level of detail, and quality of the PRA information provided in PSAR chapter 3.

To support the NRC staff's review, the NRC staff conducted an audit focused on the PRA and its supporting documentation (ML26293A340). The audit provided the NRC staff with access to PRA materials and facilitated direct discussions with the applicant to clarify technical issues related to the PRA. The NRC staff's audit included the LMGS PRA model that the applicant developed using the Computer Aided Fault Tree Analysis System (CAFTA) software. The NRC staff focused its audit of the PRA model on the model structure, development methodology, and quantification. The NRC staff also audited selected sensitivity analyses to assess the design's robustness at the CP stage and to better understand the uncertainties inherent in the PRA modeling.

As discussed below, based on the NRC staff's review of the PSAR, as confirmed in the audit, the NRC staff determined that the LMGS PRA was developed in a manner consistent with applicable guidance and in reasonable conformance with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, as endorsed by RG 1.247. The LMGS PRA demonstrates that the risk associated with the LMGS site and preliminary Xe-100 design is acceptably low and provides meaningful risk insights in support of implementing the LMP methodology described in NEI 18-04.

Additionally, the NRC staff determined that LME followed the guidance provided in RG 1.253 and addressed the eleven minimally acceptable PRA elements outlined in table A-1 at a level sufficient for the CP stage. The NRC staff's evaluation of each element is described below. LME performed a self-assessment to ensure the LMGS PRA conformed to these eleven elements in accordance with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021. LME described in PSAR section 3.1.1.1, that its self-assessment was performed following the guidance in NEI 20-09, "Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard," Revision 1 (ML20302A115), which is endorsed in RG 1.247. A summary of the NRC staff's review of LME's self-assessment is provided below.

Unlike the surrogate risk metrics commonly used in LWR PRAs, such as core damage frequency and large release frequency, the LMGS CP application employs risk metrics focused on offsite radiological consequences from ESs involving the release of radioactive material. These consequence-based metrics are aligned with those used in LWR level 3 PRAs but exclude consideration of radiological or other impacts on onsite personnel. At this stage of the design development, the applicant evaluated the following radionuclide source terms in the LMGS PRA:

- Fuel pebbles in the reactor
- Circulating activity within the helium pressure boundary (HPB)
- Radionuclides plated out on surfaces within the HPB

ASME/ANS RA-S-1.4-2021, as endorsed by RG 1.247, with exceptions, provides guidance for PRA radionuclide source terms for non-LWRs. The NRC staff reviewed the PSAR and audited supplemental materials to confirm and understand the information in the PSAR. Based on the NRC staff's review, the NRC staff determined these source terms to be reasonable and sufficient for the CP stage PRA because they were developed with approved methods and are consistent with the ASME/ANS RA-S-1.4-2021, as endorsed, with exceptions, by RG 1.247. A summary of the NRC staff's review of LME's mechanistic source term (MST) analysis is provided below in PRA element 9 and the NRC staff's review of the MST methodology is in section 3.1.4 of this SE.

PRA Element 1 – Initiating Event Analysis

As described in RG 1.247, the objective of the initiating event (IE) analysis is to identify and characterize events that could challenge safe plant operation during any POS and require effective mitigation by plant equipment and personnel to prevent or to mitigate a release of radioactive material. This includes both events that may occur at the plant and those with a credible probability of occurrence.

PSAR section 3.1.1.2 summarizes the hazards that were screened out of the PRA, the hazards that were screened into the PRA, and the hazards addressed through alternative approaches such as design-basis hazard level (DBHL) framework. Consistent with RG 1.253, section A3.3, at the CP stage some PRA elements, such as external hazards, were not completed. PSAR section 3.1.1.2 describes the internal IEs that the applicant considered for all sources of radionuclides and POSs that are identified in PSAR sections 3.1.1.2 and 3.10, and section 3.11, respectively. The LMGS IE analysis focused on internal events during full-power operation that

could impact the fuel in the reactor, circulating activity within the HPB, and radionuclides plated out on surfaces within the HPB.

To further understand the description in PSAR section 3.1.1.2 and PSAR table 3.2-2, the NRC staff audited Document ID number 000897, "Probabilistic Risk Assessment: Initiating Event Analysis – Preliminary Scoping," Revision 4. The audit focused on confirming the characteristics and attributes of the IE analysis to ensure the analysis aligns with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 and RG 1.247 and is consistent with information in the PSAR. The NRC staff examined the approach and method applied in the IE analysis, ensuring they are reasonable and consistent with accepted industry practices; the NRC staff also confirmed during the audit that the assumptions are reasonable, and the analysis aligns with the preliminary plant design to the extent possible.

Specifically, during the audit the NRC staff focused on the systematic approach used to develop a comprehensive list of potential internal IEs during full-power operation, including those affecting the reactor core, such as the steam generator (SG) tube leaks and main steam line breaks (MSLBs). The NRC staff examined how LME used multiple data sources, such as NUREG/CR-6928 (ML070650650), and relevant non-LWR and U.S. Department of Energy's standard modular high temperature gas reactor PRAs, to ensure LME's review was comprehensive, and to understand how system-level design reviews were applied to identify plant-specific initiators.

During the audit, the NRC staff also reviewed the methods used to group similar IEs and estimate their frequencies using generic data supplemented with design-specific information. The NRC staff focused on the assumptions, data sources, and treatment of parametric and modeling uncertainties in the IE analysis. The NRC staff's audit identified that the IE identification, grouping, and frequency estimation processes were conducted in a structured and traceable manner consistent with the non-LWR PRA standard. The NRC staff confirmed in the audit that the analysis includes a reasonable set of potential initiators appropriately reflects current design information and identifies key sources of uncertainty for follow-up and refinement at the OL application stage.

Based on the NRC staff's review, the NRC staff determined the IE analysis to be reasonable because the information in PSAR section 3.1.1.2, is consistent with RG 1.253 and the analysis was performed in accordance with RG 1.247, as appropriate for a preliminary design and applicable to the CP stage, as follows:

- The IE analysis includes detailed identification and characterization of IEs suitable for the design stage.
- LME logically grouped IEs based on similarity in mitigation requirements.
- LME used reasonable estimates for IE frequencies.
- LME identified and qualitatively characterized sources of model uncertainty related to the IE analysis.
- The IE analysis documentation offers traceability of the methodology employed and reasonably documents the results obtained.

As LMGS advances from preliminary design towards operation, the IE analysis and its underlying assumptions are expected to be updated and refined using increasingly detailed plant information.

PRA Element 2 - Event Sequence Analysis

As described in RG 1.247, the objective of the ES analysis is to model, in chronological order to the extent practical, the different possible potential progressions of events following the IE. These ESs lead either to successful prevention or to the release of radioactive material. The analysis will account for the availability of plant systems and operator actions based on defined success criteria (SC) and established plant operating procedures, including emergency and abnormal operating procedures.

The LMGS ES analysis, which at the CP stage is limited to full-power conditions, is described in PSAR sections 3.1.1.2 and 3.1.1.3, and PSAR table 3.2-2. PSAR section 3.1.1.3 describes that LME performed ES modeling and quantification using the Electric Power Research Institute (EPRI) Phoenix Architect, which includes CAFTA, UNCERT, FRANX, and FTREX. The ES analysis integrates POSs, IEs, safety functions, and the success or failure of SSCs, culminating in end states that may involve potential release of radioactive material. The primary output of the ES analysis is a set of event trees (ETs), which delineate possible event progressions. Each ET represents a time-independent, system-level response to a specific IE. The results provide essential inputs for plant risk quantification. PSAR table 3.2-2 provides the initiator and a brief word description of the LBE sequences.

To further understand the description in the PSAR, the NRC staff audited Document ID number 002414, “Probabilistic Risk Assessment: Event Sequence Analysis – Preliminary Scoping,” Revision 4. The audit focused on the characteristics and attributes of the ES analysis to ensure that the analysis aligns with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 and RG 1.247 and is consistent with information in the PSAR. The NRC staff examined the approaches and methodologies used in the ES analysis, ensuring that they are reasonable, technically defensible, and consistent with established industry practices. The NRC staff also confirmed in the audit, that the assumptions are reasonable and the analysis reflects a level of detail suitable for the preliminary plant design.

Specifically, during the audit, the NRC staff focused on the modeling of event progressions following the initiation of an IE for full-power conditions. The NRC staff focused on the development of ETs representing plant responses to various IEs, the identification of key radionuclide release barriers, and the integration of safety functions supporting reactivity control, primary circulator flow, core heat removal, and the HPB. The NRC staff examined how the applicant defined fuel damage, the associated end states, and the mission times applied to active and passive heat removal sequences.

The NRC staff further assessed the use of design stage assumptions, treatment of modeling and completeness uncertainties, and the documentation of SC for key safety functions. The NRC staff’s audit confirmed that the ES analysis was structured and traceable, reflected the functional dependencies influencing event progression, and followed the non-LWR PRA standard for defining barriers, SC, and mission times.

Based on the NRC staff’s review, the NRC staff determined the ES analysis to be reasonable because information in PSAR sections 3.1.1.2 and 3.1.1.3, and table 3.2-2, is consistent with

RG 1.253, and the analysis was performed in accordance with RG 1.247, as appropriate for a preliminary design and applicable at the CP stage, as follows:

- LME reasonably defined and described for each source of radioactive material and POS the barriers to radionuclide release and corresponding safety functions required to protect each barrier.
- The ES analysis reflects functional dependencies influencing event progression.
- LME considers individual function SC and associated mission times.
- LME identified and characterized relevant sources of modeling uncertainty related to the ES analysis.
- LME sufficiently documented the ES analysis to ensure traceability of the underlying technical basis.

As LMGS advances from preliminary design towards operation, the ES analysis and its supporting assumptions are expected to be updated based on the final design.

PRA Element 3 - Success Criteria Analysis

As described in RG 1.247, the objective of the SC analysis is to establish the minimum functional requirements for each key safety function and the systems that fulfill these functions to prevent or mitigate radioactive release following an IE.

To further understand LME's SC analysis, the NRC staff audited Document ID number 001994, "Xe-100 Probabilistic Risk Assessment: Success Criteria Development," Revision 3. The audit focused on the characteristics and attributes of the SC analysis to ensure the analysis aligns with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 and RG 1.247 and is consistent with information in the PSAR. The NRC staff assessed the SC analysis approach to confirm that the SC are based on reasonable engineering analyses and that the assumptions are appropriate, and the analysis aligns with the preliminary plant design.

Specifically, during the audit, the NRC staff focused on the minimum functional requirements for SSCs necessary to maintain critical safety functions during full-power conditions. The NRC staff focused on the methods and assumptions used to establish performance requirements for key safety functions, reactivity control, primary circulator flow, core heat removal, and the HPB, and the corresponding SSCs that maintain radionuclide barriers such as the HPB. The NRC staff assessed how both active and passive systems were treated in the analysis, including the determination of SC for active systems and the engineering basis for passive system performance.

The NRC staff also audited the assumptions and uncertainties in the SC analysis to confirm that they are reasonable for the CP stage, given the preliminary nature of the design. Due to the preliminary nature of the design, the NRC staff focused on understanding the treatment of modeling and completeness uncertainties and the traceability of the analytical bases supporting the SC. Overall, the NRC staff used the audit to confirm that the SC analysis appropriately

reflects the LMGS design's capability to fulfill essential safety functions and that the approach aligns with the non-LWR PRA standard for the current stage of design maturity.

Based on the NRC staff's review, the NRC staff determined the SC analysis to be reasonable because the SC analysis is consistent with RG 1.253, and the analysis was performed in accordance with RG 1.247, as appropriate for a preliminary design and applicable at the CP stage, as follows:

- The SC analysis addresses fundamental safety functions, supporting SSCs, and operator actions.
- LME identifies relevant sources of model uncertainty.
- LME documented the SC analysis to provide traceability of the technical basis at a level sufficient for the CP stage.

As LMGS advances from preliminary design towards operation, the SC analysis and its supporting assumptions are expected to be updated based on the final design.

PRA Element 4 - Systems Analysis

As described in RG 1.247, the objective of the systems analysis (SY) is to identify and model the various combinations of failures that could prevent a system from fulfilling its functions as defined by the SC. The system model incorporates hardware, instrumentation, and their associated failure modes, as well as human failure events (HFEs) that could impair system performance. Basic events representing equipment and HFEs should be developed in sufficient detail to capture dependencies among systems and to highlight specific equipment or human actions that significantly impact the system's ability to perform its intended functions.

To further understand LME's SY, the NRC staff audited the following PRA system notebooks:

- Document ID number 001998, "Probabilistic Risk Assessment: System Analysis Methodology," Revision 3;
- Document ID number 010344, "Probabilistic Risk Assessment: Main Steam and Process Steam Supply Systems – Preliminary Scoping," Revision 1;
- Document ID number 007837, "Probabilistic Risk Assessment: System Analysis – Investment Protection System and Distributed Control System," Revision 2;
- Document ID number 010343, "Probabilistic Risk Assessment: Feedwater and Condensate System – Preliminary Scoping," Revision 1;
- Document ID number 005632, "Probabilistic Risk Assessment: System Analysis: Nuclear Island Process Water System – Preliminary Scoping," Revision 2;
- Document ID number 005266, "Probabilistic Risk Assessment: Steam Generator System – Preliminary Scoping," Revision 2;

- Document ID number 002694, “Probabilistic Risk Assessment: System Analysis: Reactor Cavity Cooling System – Preliminary Scoping,” Revision 4;
- Document ID number 004705, “Probabilistic Risk Assessment: System Analysis: Conventional Island Cooling Water Supply System – Preliminary Scoping,” Revision 4;
- Document ID number 002124, “Probabilistic Risk Assessment: System Analysis: Startup and Shutdown System – Preliminary Scoping,” Revision 4;
- Document ID number 002025, “Probabilistic Risk Assessment: System Analysis: Reactor System – Preliminary Scoping,” Revision 2;
- Document ID number 002036, “Probabilistic Risk Assessment: Helium Service System (HSS) – Preliminary Scoping,” Revision 2;
- Document ID number 002035, “Probabilistic Risk Assessment: Reactor Protection System Phase 1 System Notebook – Preliminary Scoping Analysis Report,” Revision 2;
- Document ID number 001695, “Probabilistic Risk Assessment: System Analysis: Nuclear Island Cooling Water Supply System – Preliminary Scoping,” Revision 2;
- Document ID number 001697, “Probabilistic Risk Assessment: Nuclear Island HVAC System Notebook Preliminary Scoping Analysis Report,” Revision 1;
- Document ID number 001696, “Probabilistic Risk Assessment: Electrical System Notebook – Preliminary Scoping,” Revision 3;
- Document ID number 001693, “Probabilistic Risk Assessment: System Analysis: Helium Circulator System – Preliminary Scoping,” Revision 3;
- Document ID number 001580, “Probabilistic Risk Assessment: Compressed and Instrument Air System,” Revision 3;
- Document ID number 010786, “Probabilistic Risk Assessment: Risk Assessment Application Process – Reactor System Analysis,” Revision 1;
- Document ID number 010807, “Probabilistic Risk Assessment: Risk Assessment Application Process – Reactor Protection System Analysis,” Revision 1.

During the audit, the NRC staff focused on the characteristics and attributes of the SY align with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, as endorsed by RG 1.247. The NRC staff examined the approach and methods applied in the SY, ensuring they are reasonable and consistent with accepted industry practices. The NRC staff also confirmed that the assumptions are reasonable and the analysis aligns with the preliminary plant design to the extent possible.

Specifically, the NRC staff audited the modeling of system functions, dependencies, failure modes, and HFEs for key plant systems. The NRC staff focused on how each system was characterized to support the SC for essential safety functions, including reactivity control, primary circulator flow, core heat removal, and the HPB. The NRC staff assessed the inclusion of active and passive components, support systems, dependencies, mission times, component actuations, and operator actions, as well as the treatment of common cause failures and human

errors in the SY. The NRC staff also reviewed the documentation of assumptions, sources of uncertainty, and traceability of the analysis within each system notebook. The NRC staff confirmed in audit that the models, supporting data, and documentation were developed in accordance with the current design information and high level and supporting requirements of the non-LWR PRA standard. Based on the NRC staff's review, the NRC staff determined the SY to be reasonable because the analysis was performed in accordance with RG 1.247, as appropriate for a preliminary design and applicable at the CP stage, as follows:

- LME developed the system models with reasonable detail, consistent with the currently available design information.[2]
- Fault tree models reflect the plant as-designed, as-to-be-built, and as-to-be-operated within the limits of the preliminary design information available at this time.
- The fault tree models address the SC necessary to mitigate ESs.
- LME represented support systems and their dependencies within the models. Assumptions made around the availability and capability of these systems are reasonable at the CP stage. The NRC staff will review the verification of these assumptions at the OL application stage.
- The models include active and passive components, along with relevant failure modes affecting system functions.
- The models address common cause failures and human errors at a preliminary level within the systems models. The identification of human errors was limited at this stage due to incomplete information on the operation and maintenance for systems. The NRC staff will review the HFEs at the OL application stage.
- The models incorporate mission times, component actuations, functionality, and limited associated HFEs where applicable.
- LME identified and documented sources of model uncertainty related to the SY.
- The SY documentation provides traceability of the analyses performed.

As LMGS advances from preliminary design towards operation, the SY and its supporting assumptions are expected to be updated based on the final design.

PRA Element 5 - Human Reliability Analysis

As described in RG 1.247, the objective of the human reliability analysis (HRA) is to identify and evaluate HFEs that could adversely affect normal or emergency plant operations, and to quantify their associated probabilities. HFEs associated with normal operations involve actions that render systems unavailable. In contrast, HFEs related to emergency operations encompass

[2] Due to the preliminary nature of the design and unavailability of design information, some SSCs are not modeled within the CP stage PRA. The incorporation of additional SSCs as the design progresses may increase the frequency of some events because the additional SSCs may introduce new modes of failure.

human actions that, if omitted or performed incorrectly, could prevent critical systems from performing their intended safety functions.

To further understand LME's HRA, the NRC staff audited the following documents:

- Document ID number 001565, "Probabilistic Risk Assessment: Human Reliability Analysis – Preliminary Scoping," Revision 2;
- Document ID number 010773, "Probabilistic Risk Assessment: Model Gap Assessments - Human Reliability Analysis – Preliminary Scoping," Revision 2;
- Document ID number 010701, "Probabilistic Risk Assessment: Technical Adequacy Gap Assessments for the Preliminary Safety Analysis Report - Preliminary Scoping," Revision 2;
- Document ID number 010770, "Probabilistic Risk Assessment: Model Gap Assessments – System Analysis – Non-Safety-Related and Non-Safety-Related with Special Treatments – Preliminary Scoping," Revision 2.

During the audit, the NRC staff focused on the characteristics and attributes of the HRA analysis to ensure the analysis aligns with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, as endorsed by RG 1.247.

Specifically, the NRC staff focused on the identification and evaluation of HFEs that could affect normal or emergency plant operations. The NRC staff focused on the methodology used to characterize and quantify HFEs for pre-IE, IE, and post-IE actions, as well as the treatment of dependencies. The NRC staff audited the use of the HRA methodology, referenced guidance, and calculation tools to ensure the analysis aligned with the non-LWR PRA standard and systematically captured the potential impacts of human actions on system performance and plant safety. The NRC staff also examined the underlying assumptions and sources of modeling and completeness uncertainty.

Document ID numbers 001565, 010770, and 010773, describe a rudimentary HRA, which lacks a systematic process for identifying and quantifying different pre- and post-IE HFEs but relies on the operator actions and their HFEs identified in the SY and ES quantification (ESQ) analysis. At the current design stage, LME did not identify IE HFEs and did not perform a dependency analysis. The applicant did not perform a gap assessment to identify deviations from RG 1.253, appendix A. In place of the gap assessment, the applicant performed an assessment to determine the impact on the overall risk results during the CP stage and whether further detailed analysis of the HFEs is warranted for the CP application. The applicant performed sensitivity analyses to assess the impact of not meeting most of the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 high-level HRA requirements and their supporting requirements.

The applicant identified and modeled the miscalibration of reactor protection system (RPS) sensors in the PRA. The applicant assumed that any misalignments of the passive reactor cavity cooling system (RCCS) would be identified and addressed because it's part of a system that is normally in operation and therefore, did not include pre-IE HFEs for the RCCS. The NRC staff audited the applicant's assessment of the impact of the reliability of NST/NSRST SSCs, including the potential incompleteness of pre-IE HFEs as documented in Document ID number 010770. To assess the pre-IE HFEs, the applicant conducted a sensitivity analysis to determine

the impact on overall risk from failure of the NST and NSRST SSCs. The sensitivity analysis assigned LBEs the frequency of the IE associated with them, which is representative of the failure of all NST and NSRST SSCs and the success of all SR systems. The resultant mean values of frequency and consequence were compared to the F-C target in NEI 18-04. The sensitivity analysis results showed the overall change in risk for slightly less than the majority (85 percent) of the LBEs remained below the F-C target and a limited number of LBEs (six) exceeded the F-C target. Of the ones that exceeded the F-C target, none are in the PSAR LBE set, indicating that the baseline 95th percentile frequency for these events is below the beyond-design-basis events (BDBE) cutoff. This result demonstrates the level of conservatism used in this sensitivity.

To assess the post-IE HFEs, the applicant conducted a sensitivity analysis to determine the impact on overall risk from failing all currently identified post-IE HFEs and any redundant manual recovery actions. The NRC staff audited the applicant's assessment of the overall risk impact of the post-IE HFEs as documented in Document ID number 010773. The sensitivity analysis assigned all post-IE HFEs and all manual recovery actions a failure probability of 1.0, which is representative of their failure. The resultant mean values of frequency and consequence were compared to the F-C target in NEI 18-04. The sensitivity analysis results showed the overall change in risk to not be significant because the majority (greater than 93 percent) of the LBEs remained below the F-C target. This is consistent with the ASME/ANS PRA standard section 3.3.8 which states that the difference is not significant if the modeled ESs accounting for at least 90 percent of each ES family frequency associated with all the sources of radioactive material, POSs, and hazard groups being evaluated, as applicable, are not adversely affected by appropriate sensitivity studies or bounding evaluations.

The applicant's assessment concluded that the impact on the overall risk results during the CP stage is not significant and further detailed analysis of the HFEs is not warranted for the CP application.

Based on the NRC staff's review, the NRC staff observes that:

- The pre-IE HFE sensitivity analyses showed the overall change in risk to not be significant for 85 percent of LBEs and a limited overall change in risk for six LBEs that exceeded the F-C target. Based on the conservatism in this sensitivity and the limited cases that exceeded the F-C target, it is acceptable that further evaluation was not performed for the CP.

- LME's design intention of the Xe-100 reactor is to not rely on operator actions to execute the required safety functions (RSFs) and PSFs.

- The post-IE HFE sensitivity analysis showed the overall change in risk to not be significant because the majority of the LBEs remained below the F-C target, consistent with ASME/ANS PRA standard section 3.3.8.

- LME states in PSAR section 1.4, "The regulatory guidance is relevant and applicable, and the design fully conforms to the guidance," and PSAR table 1.4-2 states, "full conformance" for RG 1.247 and RG 1.253.

- LME states in PSAR section 3.1.1.1, “The LMGS PRA is developed in accordance with the requirements of the ASME/ANS Non-LWR PRA Standard.”

The NRC staff finds that the HRA used to support the PSAR does not follow the guidance in appendix A of RG 1.253, by not meeting the supporting requirements in the ASME/ANS PRA standard that RG 1.253 lists as being applicable at the CP stage. For example, the HRA lacks a systematic process for identifying and quantifying different pre- and post-IE HFEs and evaluates a limited number of human actions without consideration of dependencies. LME performed sensitivity analyses to show that even though the LMGS HRA does not meet the ASME/ANS PRA standard HRA supporting requirements listed in RG 1.253, the lack of HRA does not significantly impact the PRA model and therefore, the CP application. The NRC staff finds that the ASME/ANS PRA standard HRA technical element, which RG 1.253 indicates should be addressed as part of a minimally acceptable PRA to implement the LMP methodology at the CP stage, is not met but there is reasonable assurance that the lack of HRA is not significantly affecting the resultant PRA information presented in the PSAR. As LMGS advances from preliminary design toward operation, the HRA is expected to be developed consistent with the guidance and standard, and its supporting assumptions are expected to be updated based on the final design.

PRA Element 6 - Data Analysis

As described in RG 1.247, the objective of the data analysis (DA) is to quantify the failure probabilities and unavailability’s of the modeled SSCs by:

- Defining parameter boundaries,
- Grouping components appropriately,
- Ensuring consistency between parameter data and their definitions,
- Incorporating relevant generic industry data, design-specific information, and plant-specific evidence in parameter estimation, and
- Addressing the associated parameter uncertainties.

To further understand LME’s data analysis, the NRC staff audited Document ID number 002622, “Probabilistic Risk Assessment: Data Analysis – Preliminary Scoping Analysis Report,” Revision 3. During the audit, the NRC staff focused on the characteristics and attributes of the DA to ensure the analysis aligns with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, as endorsed by RG 1.247.

Specifically, the NRC staff focused on the failure probabilities and unavailabilities of the SSCs modeled in the PRA. The NRC staff focused on the methods used to estimate demand-based and time-based failure rates, including the use of industry data from NUREG/CR-6928 and other references, incorporation of design-specific information, and treatment of parametric uncertainty using beta and gamma distributions. The NRC staff also examined how the applicant defined component boundaries; grouped components; represented dependencies, including plant-level, intersystem, intra-system, and human actions; and modeled common cause failures.

The NRC staff also audited the traceability of parameter definitions to the PRA logic model and the identification and characterization of uncertainty. The NRC staff determined the assumptions and sources of modeling and completeness uncertainties to be reasonable for the current design stage.

Based on the NRC staff's review, the NRC staff determined the DA to be reasonable because the analysis was performed in accordance with RG 1.247, as appropriate for a preliminary design and applicable at the CP stage, as follows:

- Parameter estimation for basic event failure probabilities and unavailabilities incorporated generic and design-specific data. The NRC staff will review the changes made to the basic event probabilities and unavailabilities from the CP application to those presented in the OL application and the bases for the change.
- The applicant defined each parameter with respect to the PRA logic model and probability evaluation framework.
- The applicant established component boundaries that support parameter estimation.
- The applicant identified and characterized uncertainty in the parameter estimates.
- The applicant sufficiently documented the DA for the CP stage. The NRC staff will confirm that the applicant conforms with the non-LWR PRA standard, as endorsed by RG 1.247, at the OL application stage. This review will ensure traceability to the underlying technical basis at the OL application stage.

As LMGS advances from preliminary design towards operation, the DA and its supporting assumptions are expected to be updated based on the final design.

PRA Element 7 - Hazard Screening

As described in RG 1.247, the objective of the hazards screening (HS) analysis is to systematically identify all natural and human-induced hazards and, where applicable, provide a technically justified basis for the exclusion of specific hazards or hazard groups from further PRA consideration. Screening methods are typically used to demonstrate that the potential contribution of a hazard to key risk metrics, such as radiological dose or public health effects, is negligible.

PSAR section 3.1.1.2 describes the results of the HS analysis performed for the CP application. To further understand the description in the PSAR, the NRC staff audited the following documents:

- Document ID number 009284, "Probabilistic Risk Assessment: Other Hazards and Screening – Preliminary Scoping," Revision 1
- Document ID number 010791, "Probabilistic Risk Assessment: Model Gap Assessments - Other Hazards and Screening – Preliminary Scoping," Revision 1

During the audit, the NRC staff focused on the characteristics and attributes of the HS analysis to ensure the analysis aligns with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, as

endorsed by RG 1.247. LME identified potential hazards and evaluated each for potential inclusion in the PRA. LME stated that the hazards were assessed using the qualitative or quantitative screening criteria outlined in the non-LWR PRA standard ASME/ANS RA-S-1.4-2021.

The NRC staff focused on the identification and evaluation of the potential natural and human-induced hazards for inclusion in the PRA. The NRC staff focused on assessing the methods used for qualitative screening to ensure consistency with the non-LWR PRA standard. The audit confirmed that hazards were screened out if bounding evaluations demonstrated a negligible impact on plant risk, while those retained were subject to further qualitative analysis, quantitative analysis, or included within the PRA scope. The NRC staff's audit confirmed the assumptions underlying the HS analysis, as well as the identified sources of modeling and completeness uncertainties, are reasonable for the current design stage.

Based on the NRC staff's review, the NRC staff determined the HS analysis to be reasonable because information in PSAR section 3.1.1.2, pertinent to the analysis, is consistent with RG 1.253 guidance for CP applications and the analysis was performed in accordance with RG 1.247 as appropriate for a preliminary design and applicable at the CP stage, as follows:

- The applicant systematically identified natural and human-induced hazards that could affect the design, plant, or site, including site- and design-specific hazards.
- The applicant conducted initial screening using reasonably defined qualitative criteria.
- Quantitative screening applied bounding assessments using predefined criteria.
- The applicant evaluated the available data supporting the quantitative screening decisions.
- The applicant identified and characterized uncertainties associated with the screening process.
- The applicant documented the HS analysis in detail to ensure traceability of the assessment.

LME is expected to reassess the HS assumptions, sources of modeling, and completeness uncertainty at the OL application stage, based on final design information, and either confirm its continued validity or update the HS analysis as necessary.

PRA Element 8 - Event Sequence Quantification Analysis

As described in RG 1.247, the objective of the ESQ analysis is to estimate the frequencies of ESs and ES families across all phases of the plant lifecycle. This ensures that all risk-significant contributors are identified, characterized, and well understood. The ESQ analysis addresses key dependencies and provides a robust evaluation of the uncertainties and assumptions inherent in the PRA, including their impact on overall risk insights.

PSAR section 3.1.1.6 describes the CP application's ESQ analysis. To further understand the description in the PSAR, the NRC staff conducted an audit that included Document ID number 005147, "Probabilistic Risk Assessment: Event Sequence Quantification Preliminary Scoping

Analysis Report," Revision 6. The audit focused on the characteristics and attributes of the ESQ analysis to ensure the analysis aligns with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021. The ESQ analysis integrates ESs, system models, HRA, and data to quantify the frequencies of each modeled ES and ES family. These quantified results serve as inputs to risk integration (RI) tasks, supporting evaluations against QHO risk metrics.

Individual sequence results were aggregated for reporting and served as input for the LMP risk evaluation. Quantification results are expressed in terms of minimal cutsets representing the smallest combinations of failures or conditions that lead to radionuclide release.

The NRC staff focused on the integration of ESs, system models, HRA, and data to quantify the frequencies of ESs across the plant lifecycle. The NRC staff focused on the methodology for generating minimal cutsets, the use of CAFTA, PRAQuant, cQuant, UNCERT, ACUBE, and FTREX for modeling and quantification, and the verification of cutsets to ensure proper integration of ETs, fault trees, system models, and human actions. The NRC staff also assessed the application of truncation limits, the aggregation of sequence results for release categories, and the derivation of equipment and operator importance measures.

The NRC staff's audit confirmed that the LMGS ESQ analysis reasonably incorporates dependencies, functional interactions, and uncertainties, while identifying and characterizing risk-significant contributors. The NRC staff finds that the assumptions underlying the ESQ analysis are reasonable for the current design stage.

Based on the NRC staff's review, the NRC staff determined the ESQ analysis to be reasonable because information in PSAR section 3.1.1.6, is consistent with RG 1.253 and the analysis was performed in accordance with RG 1.247, as appropriate for a preliminary design and applicable at the CP stage, as follows:

- The LMGS ESQ analysis integrated all relevant modeling elements.
- LME quantified ESs using suitable models and software, with sufficiently low truncation limits to ensure convergence of risk results.
- LME identified and characterized the risk-significant contributors to each sequence.
- LME reasonably addressed and quantified the uncertainties in the quantification process.
- LME sufficiently documented the ESQ analysis in sufficient detail to ensure traceability of the assessment performed.

Due to the preliminary stage of the LMGS design in the CP application, LME is expected to reassess the ESQ assumptions at the OL application stage, based on finalized design information, and either confirm their continued validity or make appropriate updates to the ESQ analysis.

Additionally, since ESQ analysis involves several aspects related to event frequency, its acceptability is coupled to the acceptability of the other elements (e.g., IE, HRA, SY, ES).

PRA Element 9 - Mechanistic Source Term Analysis

As described in RG 1.247, the objective of the MST analysis PRA element (MS) is to characterize potential radiological releases to the environment resulting from ESs that could lead to radionuclide release. This characterization includes identifying risk-significant isotopes for consequence assessment and providing key input parameters such as release locations, the physical and chemical forms of radionuclides, time-dependent isotopic release rates to the atmosphere, the heat content of the carrier fluid, and parameters necessary to estimate plume buoyancy. The MS analysis is expected to produce MSTs suitable for use in radiological consequence evaluations.

The MST analysis characterizes the source term release based on the status of the HPB and the reactor building (RB). The overall source term is associated with one of four release states: no release from intact HPB, release of circulating activity only, release of circulating activity and lift off, or delayed fuel release. Each unique source term is made up of a combination of the HPB status, RB status, source term category and a source term modifier to account for unique characteristics that could impact the overall source term release and timing. The MST analysis models the following release mechanisms: fuel heat up, steady-state inventory, lift off, wash off, steam off, and long-term release. It also addresses radionuclide transport within the plant and potential environmental release pathways.

Multiple computational tools were employed to develop the source term estimates, including GOTHIC, Flownex, and the XSTERM code suite. Bounding source terms were calculated to account for uncertainties inherent in the source term estimation process. Section 3.1.4 of this SE provides a detailed review of the MST methodology, which is provided in PSAR section 3.1.4.

To further understand LME's MS analysis, the NRC staff audited Document ID number 006869, "Probabilistic Risk Assessment: Mechanistic Source Term Analysis – Preliminary Scoping," Revision 3, including the assumptions supporting the MS analysis and their associated sources of modeling and completeness uncertainty. The NRC staff assessed the potential radiological releases from ESs. The NRC staff reviewed the radionuclide transport within the plant and potential environmental release pathways.

The NRC staff confirmed that the MS analysis reasonably applies appropriate methodologies and computational tools, and identifies key uncertainties associated with release mechanisms and transport phenomena. The NRC staff finds that the assumptions supporting the analysis and associated uncertainties, are reasonable for the current design stage.

Based on the NRC staff's review, the NRC staff determined the MS analysis to be reasonable because information in PSAR section 3.1.4 is consistent with RG 1.253, and the analysis was performed in accordance with RG 1.247, as appropriate for a preliminary design and applicable at the CP stage, as follows:

- LME reasonably grouped the radionuclide releases into representative source terms or release categories.
- LME assessed each release category with respect to timing, location, magnitude, transport barriers, and mechanisms of release and transport.

- LME developed radiological source terms using methodologies and computational tools to the extent practicable.
- LME identified and characterized the uncertainties in the MSTs and associated transport phenomena.
- LME documented the MS analysis to ensure traceability of the technical basis.

As LMGS advances from preliminary design towards operation, the MS analysis and its supporting assumptions are expected to be updated based on the final design.

PRA Element 10 - Radiological Consequence Analysis

As described in RG 1.247, the objective of the radiological consequence analysis is to evaluate the potential impacts of radioactive material releases from a nuclear facility under various operational and accident conditions.

To further understand LME's radiological consequence (RC) analysis, the NRC staff audited Document ID numbers 006870, "Probabilistic Risk Assessment: Radiological Consequence Analysis – Preliminary Scoping," Revision 5 and 002200, "Probabilistic Risk Assessment: Risk Integration – Preliminary Scoping," Revision 5. The audit focused on the characteristics and attributes of the RC analysis to ensure the analysis aligns with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 and RG 1.247, and is consistent with information in the PSAR.

The RC analysis used source terms developed through the MS analysis as input. The RC analysis quantified individual risk by integrating health effects and dose exceedance probabilities with the annual frequencies of each source and release category. LME used the XDIS module of the XSTERM code to perform the RC calculations.

The radiological consequences evaluated for each source term release category include:

- Total effective dose equivalent (TEDE) to a receptor located at the control room, EAB and beyond EAB.
- Frequency of exceeding 100 mrem (1 mSv) TEDE at the site boundary.
- Average individual early fatality risk within one mile of the EAB.
- Average individual latent cancer fatality risk within ten miles of the EAB.

LME used Flownex and XSTERM to perform individual calculations, XSTERM to calculate the radionuclide inventory released, and then the XDIS module to calculate the dose. LME used XSTERM to calculate the dose associated with the release of the steady-state circulating helium inventory. LME used XSTERM which uses Flownex Screening Criteria (FSC) based on Flownex calculated fuel pebble temperatures to calculate the dose associated with radionuclides released from the fuel during a fuel heat-up transient. During depressurization events, LME considered dose from liftoff of previously deposited radionuclides within the HPB and given a steam generator tube rupture (SGTR) event, LME considered dose from steamoff and washoff. LME used Phenomena and Key Parameter Identification and Ranking Table (PKPIRT) design-basis accident (DBA) to develop the uncertainties associated with TEDE. LME developed mean

values and 95th percentile estimates, based on analyses of nominal and bounding source terms to ensure a robust evaluation. The NRC staff's evaluation of the dose methodologies used in RC analysis are presented in section 3.1.5 of this SE.

The NRC staff focused on the evaluation of potential impacts of radioactive material releases under various operational and accident conditions. The NRC staff examined the RC analysis, which incorporates source terms from the MS analysis, site-specific characteristics, meteorology, atmospheric transport and dispersion, protective actions, dosimetry calculations, and health effects estimation. The NRC staff reviewed the calculations performed to quantify TEDE at the control room, EAB, and beyond EAB. The NRC staff also reviewed the mean and 95th percentile estimates that were developed to account for uncertainties associated with the source terms. The NRC staff determined that the assumptions relevant to the RC analysis are reasonable at the CP stage.

Based on the NRC staff's review, the NRC staff determined the RC analysis to be reasonable because information in the PSAR and Document ID numbers 006870 and 002200, is consistent with RG 1.253 and the analysis was performed in accordance with RG 1.247, as appropriate for a preliminary design and applicable at the CP stage, as follows:

- LME used methodologies and computational tools that have been evaluated at a preliminary level and have verification and validation (V&V) plans in place that will be executed prior to the OL application, to conduct the RC analysis.
- LME identified, characterized, and quantified to the extent practicable the uncertainties associated with TEDE and associated risk metrics.
- LME documented the RC analysis to provide traceability of the technical basis at a level sufficient for the CP stage.

LME is expected to reassess RC analysis assumptions at the OL application stage using final design information and either confirm their continued validity or revise the RC analysis accordingly.

PRA Element 11 - Risk Integration

As described in RG 1.247, the objectives of the RI analysis are to: (1) establish criteria for determining the risk significance of ESs; (2) quantify overall plant risk using appropriate metrics; and (3) characterize and quantify uncertainties associated with those metrics.

PSAR sections 3.1.1.7, 3.2.2, and 4.1.1, summarize the results of the CP application's RI analysis. To further understand LME's RI analysis, the NRC staff audited Document ID number 002200, "Probabilistic Risk Assessment: Risk Integration – Preliminary Scoping," Revision 5. The audit focused on the characteristics and attributes of the RI analysis to ensure that the analysis aligns with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 and RG 1.247, and is consistent with information in the PSAR.

Specifically, the NRC staff focused on the integration of the ESQ and RC analysis into an overall integrated risk analysis. Other internal and external hazards (e.g., internal and external fires, internal and external flooding, seismic events, high winds, tornadoes) were not included in this integration, which is consistent with appendix A of RG 1.253. The NRC staff confirmed that

the integrated model was evaluated against the plant-level cumulative risk metrics summarized in PSAR section 4.1.1, and verified the application of truncation thresholds to ensure adequate resolution of low-frequency contributors. The NRC staff's evaluation regarding the LMGS integrated risk results is documented in chapter 4 of this SE.

Based on the NRC staff's review, the NRC staff determined the RI analysis to be reasonable because information in PSAR sections 3.1.1.7, 3.2.2 and 4.1.1, is consistent with RG 1.253, including that external hazards such as internal fires, internal flooding, seismic events, or high winds were not included, and the analysis was performed in accordance with RG 1.247, as appropriate for a preliminary design and applicable at the CP stage, as follows:

- LME identified for each PRA element, the assumptions and the sources of uncertainty and evaluated at a preliminary level their potential impacts on ES frequencies and consequence estimates were.

- Because of limited availability of design and site information, LME used simplified methods to evaluate the integrated risk, which considered the hazards and ESs within the scope of the PRA consistent with the design at the CP stage. The NRC staff finds the simplified methods of calculating acute deaths and latent cancer fatalities to assess the QHOs based on limited site information at the CP stage to be reasonable. The assessment of the QHOs using site information and approved methods is required at the OL application stage, when site information becomes available, in accordance with RG 1.253.

- LME characterized or quantified uncertainty. The NRC staff recognizes that uncertainties related to portions of the design not modeled at the CP stage will be assessed at the OL application stage, when design information becomes available, in accordance with RG 1.253.

- LME documented the RI analysis to provide traceability of the technical basis and supporting assumptions at a level sufficient for the CP stage.

As LMGS advances from preliminary design towards operation, the RI analysis and its supporting assumptions are expected to be updated based on the final design.

PRA Self-assessment

The LMGS PRA self-assessment is described in PSAR section 3.1.1.1, which states, "The PRA self-assessment evaluated the PRA's conformance to the ASME/ANS non-LWR PRA standard technical requirements, as endorsed by trial use RG 1.247, that are applicable. This assessment included the technical requirements in RG 1.253 Table A-2." The eleven elements applicable are also the subject of corresponding regulatory positions in RG 1.247.

To further understand the description in the PSAR, the NRC staff conducted an audit that included:

- Document ID number 010791, "Probabilistic Risk Assessment: Model Gap Assessments - Other Hazards and Screening – Preliminary Scoping," Revision 1.

- Document ID number 010773, “Probabilistic Risk Assessment: Model Gap Assessments - Human Reliability Analysis – Preliminary Scoping,” Revision 2.
- Document ID number 010770, “Probabilistic Risk Assessment: Model Gap Assessments – System Analysis – Non-Safety-Related and Non-Safety-Related with Special Treatment Systems - Preliminary Scoping,” Revision 2.
- Document ID number 010701, “Probabilistic Risk Assessment: Technical Adequacy Gap Assessments for the Preliminary Safety Analysis Report - Preliminary Scoping,” Revision 2.
- Document ID number 010771, “Probabilistic Risk Assessment: Model Gap Assessment – Initiating Event Analysis - Preliminary Scoping,” Revision 2.
- Document ID number 010344, “Probabilistic Risk Assessment: Main Steam and Process Steam Supply Systems – Preliminary Scoping,” Revision 1.
- Document ID number 010343, “Probabilistic Risk Assessment: Feedwater and Condensate System – Preliminary Scoping,” Revision 1.
- Document ID number 007837, “Probabilistic Risk Assessment: System Analysis – Investment Protection System and Distributed Control System,” Revision 2.
- Document ID number 006939, “Probabilistic Risk Assessment: Technical Adequacy Summary for the Preliminary Safety Analysis Report,” Revision 1.
- Document ID number 005632, “Probabilistic Risk Assessment: System Analysis: Nuclear Island Process Water System – Preliminary Scoping,” Revision 2.
- Document ID number 005266, “Probabilistic Risk Assessment: Steam Generator System – Preliminary Scoping,” Revision 2.
- Document ID number 002694, “Probabilistic Risk Assessment: System Analysis: Reactor Cavity Cooling System – Preliminary Scoping,” Revision 4.
- Document ID number 004705, “Probabilistic Risk Assessment: System Analysis: Conventional Island Cooling Water Supply System – Preliminary Scoping,” Revision 4.
- Document ID number 002124, “Probabilistic Risk Assessment: System Analysis: Startup and Shutdown System – Preliminary Scoping,” Revision 4.
- Document ID number 002025, “Probabilistic Risk Assessment: System Analysis: Reactor System – Preliminary Scoping,” Revision 2.
- Document ID number 002036, “Probabilistic Risk Assessment: Helium Service System (HSS) – Preliminary Scoping,” Revision 2.
- Document ID number 002035, “Probabilistic Risk Assessment: Reactor Protection System Phase 1 System Notebook – Preliminary Scoping Analysis Report,” Revision 2.
- Document ID number 001695, “Probabilistic Risk Assessment: System Analysis: Nuclear Island Cooling Water Supply System – Preliminary Scoping,” Revision 2.

- Document ID number 001697, “Probabilistic Risk Assessment: Nuclear Island HVAC System Notebook Preliminary Scoping Analysis Report,” Revision 1.
- Document ID number 001696, “Probabilistic Risk Assessment: Electrical System Notebook – Preliminary Scoping,” Revision 3.
- Document ID number 001693, “Probabilistic Risk Assessment: System Analysis: Helium Circulator System – Preliminary Scoping,” Revision 3.
- Document ID number 001580, “Probabilistic Risk Assessment: Compressed and Instrument Air System,” Revision 3.
- Document ID number 010786, “Probabilistic Risk Assessment: Risk Assessment Application Process – Reactor System Analysis,” Revision 1.
- Document ID number 010803, “Probabilistic Risk Assessment: Model Gap Assessment – Success Criteria Development – Preliminary Scoping,” Revision 1.
- Document ID number 010807, “Probabilistic Risk Assessment: Risk Assessment Application Process – Reactor Protection System Analysis,” Revision 1.
- Document ID number 010779, “Probabilistic Risk Assessment: Technical Adequacy Gap Assessments – Event Sequence Analysis - Preliminary Scoping,” Revision 1.
- Document ID number 010780, “Probabilistic Risk Assessment: Technical Adequacy Gap Assessments – Risk Integration - Preliminary Scoping,” Revision 1.

The audit focused on confirming that the self-assessment was performed in accordance with RG 1.247 and RG 1.253, as provided in PSAR section 3.1.1.1. The NRC staff made several observations regarding the self-assessment including the following:

- The scope of the self-assessment was consistent with the guidance in RG 1.253, including the eleven elements listed above in this SE. Consistent with RG 1.253, the following PRA elements were not analyzed for the CP application and were therefore excluded from the LMGS PRA self-assessment: POS Analysis, internal flood, internal fire, seismic, high wind, external flooding and other external hazards PRAs.
- The assessment results were documented and made available for the NRC staff’s audit.
- The self-assessment appeared to be conducted primarily as individual comparisons of PRA standard supporting requirements against documentation. Enhanced team-based holistic evaluation of the adequacy of the PRA models and the underlying technical assessments may result in enhanced coordination on resolution plans to address identified self-assessment findings.
- Portions of the self-assessment appeared to have been performed by various individuals and the documentation did not include their qualifications. Limited PRA expertise could lead to varying interpretations of the PRA supporting requirements and potentially influence the overall conclusions and grading outcomes.

- While recognizing that the LMGS PRA self-assessment serves as a valuable initial tool for evaluating PRA quality, the NRC staff does not rely solely on it to determine PRA adequacy or acceptability, even for PRA supporting requirements assessed as "Met."

The NRC staff audited the grading of PRA supporting requirements, identification of gaps, and the documentation of assumptions to ensure that the self-assessment appropriately captured areas where PRA supporting requirements were categorized as "Met," "Met CC-I," "Partially Met," "Not Met," or "Not Applicable." The NRC staff's audit focused on understanding the preliminary nature of the design information and PRA, and the extent to which assumptions, uncertainties, and gaps were identified and characterized to support future resolutions.

The NRC staff also audited the gap resolution, potential impacts on risk-informed applications, and verification of configuration control practices. While acknowledging limitations in the self-assessment, such as its individual-based approach and the variable level of PRA expertise involved, the self-assessment appeared to be systematic, sufficiently documented, and provided useful insights to support future refinement and verification of the PRA.

Based on the NRC staff's review of the PSAR and confirmation via the audit described above, the NRC staff finds that the LMGS PRA self-assessment was conducted per RG 1.247 and RG 1.253. In PSAR section 3.1.1.1, LME stated, "While the LMGS PRA has not been peer reviewed in accordance with the NEI 20-09 process, a PRA peer review will be completed to support the [OL] application…."

PRA Configuration Control Program

PSAR section 3.1.1.1 describes the LMGS PRA configuration control process. The non-LWR PRA standard ASME/ANS RA-S-1.4-2021 describes that the PRA results used to support licensing or risk-informed applications, including LMP, must be derived from a PRA model that represents the as-designed, as-to-be-built, and as-to-be-operated or the as-built and as-operated plant. The PRA model's level of detail should be commensurate with the intended application and its associated decision-making requirements. Consequently, as discussed in RG 1.247 and RG 1.253, the design-specific PRA should be maintained and updated as necessary to ensure ongoing consistency with the plant's design and operational status throughout the various licensing stages, in alignment with the availability of plant information.

Accordingly, LME has established its PRA configuration control procedure, documented in Document ID number 009473, "Probabilistic Risk Assessment: Configuration Control Program," Revision 2, which defines the process for maintaining and managing the LMGS PRA model, its documentation, model change tracking, and supporting information.

The NRC staff's audit of the LMGS PRA configuration control program confirmed its ability to effectively maintain and manage the PRA model and its supporting documentation throughout the plant's design and operational lifecycle. The audit focused on the processes for:

- monitoring changes in plant design, operations, PRA technology, and industry experience;

- maintaining alignment of the PRA model with the as-designed, as-to-be-built, and as-to-be-operated plant; and

- systematically evaluating the cumulative impact of pending changes on risk-informed applications.

The NRC staff also examined how configuration control is applied to computer codes and associated files used for PRA quantification, and how updates and modifications are documented to ensure traceability. The NRC staff's audit included the review of LME's design change control and corrective action programs used to track design modifications, discrepancies between the plant and the PRA model, and relevant operating experience. The NRC staff's audit confirmed that the LMGS PRA configuration control procedure is reasonably structured and maintains effective controls over PRA model changes. The NRC staff notes that LME's configuration control program lacks defined criteria and triggers for model updates and states that this is because the design of the Xe-100 actively being developed and the PRA model is actively being revised to risk-inform the design.

Based on the NRC staff's confirmation via its audit of Document ID number 009473, the NRC staff determined for the CP application, that:

- The LMGS PRA configuration control procedure is reasonably structured to monitor PRA inputs and support the integration of new information that may influence the PRA.
- The cumulative impact of pending plant changes will be systematically evaluated.
- Configuration control over computer codes and associated files used for PRA quantification is maintained.
- The process includes updating the PRA model in response to new information.

Overall, the NRC staff determined the LMGS PRA configuration control procedure to be reasonably developed. It establishes effective processes and controls to maintain and enhance the PRA as the plant design evolves, thereby ensuring sustained support for LMP and other risk-informed initiatives.

Assessment of Conformance with the NRC Staff's Positions in RG 1.253 Regarding PRA and Its Results

The NRC staff also evaluated the LMGS PRA and its supporting PRA-related documentation available through the PRA audit, as mentioned above, against the applicable regulatory positions outlined in RG 1.253, which describes an approach that is acceptable to the NRC staff for using a technology-inclusive content-of-application methodology to inform the PSAR. The NRC staff's determinations are summarized as follows:

1. RG 1.253, section A.2.2, states that the CP applicant should clearly document, in the PSAR, the key assumptions made during the development of the LMP-based safety analysis and in selecting PRA model elements.

   The NRC Staff's Determination: The NRC staff determined the regulatory position in RG 1.253, section A.2.2 is satisfied because PSAR 3.1.1.1 identifies the key assumptions used in the PRA development.

2. RG 1.253, section A.2.3, states that the CP applicant considers both near-term and long-term uses of the PRA during its development to ensure continued applicability. Besides supporting LMP implementation, PRA results may be utilized to demonstrate regulatory compliance and support other voluntary risk-informed applications.

   The NRC Staff's Determination: Beyond supporting the LMP methodology, the LMGS PRA informs other activities that are described in the LMGS CP application, such as the design reliability assurance program, reliability and integrity management program, and environmental reviews.

   Since the LMGS CP PRA only addresses internal events and relies on key assumptions not yet validated at the CP stage due to the preliminary nature of the design, its acceptability to support full implementation of all cited intended uses is not yet determined. However, based on the evaluations in the previous subsections of SE section 3.1.1.1, the NRC staff determined that the methodologies and processes used in developing the internal events PRA portion is generally reasonable and consistent with RG 1.247 guidance for the CP stage.

3. RG 1.253, section A.3.1, states that a CP applicant using the LMP methodology should estimate the risk metrics described in sections C.3.2.1, C.4.1.1, C.4.1.2, and C.4.1.3 of 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), on either a qualitative or quantitative basis consistent with the information available when the CP application is prepared. Such an estimation should include an explanation of how the Commission's QHOs from the Commission Safety Goal Policy Statement will be met in support of the OL application.

   The NRC Staff's Determination: PSAR section 3.2 provides LME's summary evaluation of LBEs against the F-C target, consistent with NEI 21-07 section C.3.2.1, with further detail on each LBE provided in PSAR sections 3.3 through 3.5. The NRC staff's evaluation of the LBE summary and detailed evaluation of LBEs is provided in SE sections 3.3 through 3.5.

   PSAR section 4.1 provides the three cumulative plant risk performance metrics, including: 1) site boundary risk, 2) early fatality risk, and 3) latent cancer risk. The NRC staff's evaluation of these risk metrics is provided in chapter 4.1 of this SE. Based on the NRC staff's evaluation of the calculated LMGS LBEs evaluation against the F-C target and cumulative risk metrics, the NRC staff determined that regulatory position A.3.1 in RG 1.253, is fulfilled.

4. RG 1.253, section A.3.2, states that the CP applicant should:

   a. Identify all radiological sources, hazards, and POSs by performing a comprehensive and systematic search.

   b. Disposition the search results by a combination of PRA logic modeling, acceptable screening methods, risk-informed supplemental evaluations, and crediting DBHLs.

   The NRC Staff's Determination: PSAR section 3.1.1.2 identifies the radiological sources evaluated in the PRA. PSAR section 3.1.1.2 summarizes the hazards that were addressed by detailed PRA or DBHLs. PSAR section 3.1.1.2 lists the POSs that were

included in the scope of the LMGS PRA. Based on the evaluation in the previous subsections of SE section 3.1.1.1 pertaining to source, hazard, and POS identification, the NRC staff determined that the applicant satisfies the regulatory position in item (a) above.

In addition to the LMGS PRA development, LME's risk-informed supplemental evaluations are documented in PSAR sections 3.10 and 3.11, and the LME assessment of the site-specific DBHLs is documented in PSAR chapter 2, "Site Characterisitics," and PSAR section 6.1.1, thereby the applicant addressed the regulatory position in item (b) above.

5. RG 1.253, section A.3.3, states that the minimum scope of the CP stage PRA logic model should include the internal events hazard group for the reactor in the at-power POS.

   The NRC Staff's Determination: The LMGS CP stage PRA addresses internal events by including all eleven internal event elements identified in RG 1.253, table A-1. The PRA scope aligns with RG 1.253 guidance, with any deviations appropriately documented and justified as discussed above in this SE. Therefore, the NRC staff determined that the applicant fully satisfies the regulatory position of RG 1.253, section A.3.3.

6. RG 1.253, section A.3.4 states that the PRA high-level requirements[3] (HLRs) and associated supporting requirements from the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, as endorsed in RG 1.247, provide an acceptable approach for PRA logic model development.

   The NRC Staff's Determination: LME developed the CP stage PRA in accordance with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 and conducted a PRA self-assessment against the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 HLRs and supporting requirements, as discussed in PSAR section 3.1.1.1. For those PRA supporting requirements identified as "Not Met" or "Partially Met," LME provided justifications consistent with the standard guidance. The NRC staff determined that LME has satisfactorily met the regulatory position of RG 1.253, section A.3.4.

7. RG 1.253, section A.3.5, states that section 4.3.11 of the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, provides an acceptable approach for performing HS analyses.

   The NRC Staff's Determination: The HS analysis was performed and documented in PSAR section 3.1.1.2. The NRC staff verified that LME followed the guidance in the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 when conducting the HS. The NRC staff determined that LME's HS approach is consistent with the regulatory position in RG 1.253, section A.3.5, for a CP stage PRA.

[3] RG 1.253, section A.3.4, also includes a footnote that states, "The non-LWR PRA standard uses the terms 'requirement,' 'require,' and other similar mandatory language. However, the use of this language in this RG does not imply that this RG imposes any regulatory requirement or suggest that these standards are the only way to meet the statutory and regulatory requirements."

8. RG 1.253, section A.3.6, states that non-LWR CP applicants utilizing NUREG-1855 guidance to develop risk-informed supplemental evaluations should: (1) describe and justify the use of reactor-technology-specific screening criteria, and (2) explain how specific sources of uncertainty were identified and addressed.

   The NRC Staff's Determination: The risk-informed supplemental evaluations were performed and documented in PSAR section 3.10. The NRC staff verified that LME followed the guidance in NUREG-1855, by providing justifications for the reactor-technology-specific screening criteria in the risk-informed supplemental evaluations. The NRC staff determined that LME's use of the NUREG-1855 guidance approach for the reactor-technology-specific screening criteria at the CP stage PRA, is consistent with the regulatory position in RG 1.253, section A.3.6. However, the specific sources of uncertainty for the risk-informed supplemental evaluations were not identified and addressed by LME at the CP stage. LME established an upper bound on the radiological consequences for events involving a single pebble by assuming that the source term was comprised of the entire radionuclide inventory of the pebble and that release occurred all at one time in the risk-informed supplemental evaluations. An additional analysis for a large multi-pebble release used conservative assumptions to try and bound other potential release scenarios. Based on the "bounding" and "conservative, but not bounding" approaches used which are consistent with NUREG-1855, the NRC staff finds the identification and addressing of sources of uncertainty to be of a small overall facility risk.

9. RG 1.253, section A.3.7, states that CP applicants may disposition certain hazards by crediting DBHLs instead of explicitly modeling these hazards in the PRA or addressing them through risk-informed supplemental evaluations.

   The NRC Staff's Determination: In response to the NRC staff audit questions, LME updated PSAR section 3.1.1 to state that the DBHLs corresponding to deterministic external hazards are described in PSAR chapter 2 and PSAR section 6.1.1 and DBHLs corresponding to internal hazards are described in PSAR section 6.1.1 and that LME intends to use ASME/ANS RA-S-1.4-2021, as endorsed by RG 1.247, to assess the internal hazards and their corresponding DBHLs in support of the OL application submission. The NRC staff determined that LME's approach aligns with the regulatory position in RG 1.253, section A.3.7.

10. RG 1.253, section A.4.1, identifies the minimally acceptable PRA elements and additional elements recommended for full LMP methodology implementation at the CP stage.

    The NRC Staff's Determination: LME addressed all minimally acceptable PRA elements as previously described in section 3.1.1 of this SE, under each PRA element. Accordingly, the NRC staff determined that LME has met the regulatory position of RG 1.253, section A.4.1.

11. RG 1.253, section A.4.2, states that PRA elements should be developed consistently with the PRA HLRs and associated supporting requirements of the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, with all HLRs expected to be met.

The NRC Staff's Determination: The NRC staff determined that the regulatory position in RG 1.253, section A.4.2, is satisfied because LME addressed all HLRs applicable at the CP stage through its self-assessment and provided justifications for any gaps, primarily due to undeveloped areas such as operator and maintenance procedures, and explained its impacts on PRA applications in their PRA self-assessment reports.

12. RG 1.253, section A.5.1, states that the CP stage PRA's level of detail should be commensurate with the preliminary plant design and site characteristics described in the PSAR.

    The NRC Staff's Determination: The level of detail in the PRA used to support the CP is consistent with the preliminary design described in other sections of the PSAR. While the PRA incorporates many assumptions about some aspects of plant operation, they are generally reasonable given the design maturity at the time of the CP application. The NRC staff determined that LMGS's PRA development satisfies the regulatory position in RG 1.253, section A.5.1, utilizing available design information as previously described in section 3.1.1 of this SE, under each PRA element.

13. RG 1.253, section A.5.2, states, in part, that when certain PRA elements cannot be met due to the maturity of preliminary plant design and information about site characteristics, the application should justify the adequacy of the internal events PRA logic model for the reactor in the at-power POSs to support the implementation of the LMP methodology.

    The NRC Staff's Determination: LME acknowledged gaps and limitations in the LMGS PRA and will develop a PRA for the OL application stage that comprehensively addresses all radiological sources, POSs, and hazards in accordance with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, as discussed in PSAR section 3.1.1. The LMGS CP stage PRA self-assessment provides justifications regarding the impact of current gaps on LMP implementation. Although some justifications cannot yet be confirmed due to incomplete design information, which is expected due to the preliminary design available at the CP stage, the NRC staff determined that LME has met the regulatory position in RG 1.253, section A.5.2, because it provided justification on PRA adequacy.

14. RG 1.253, section A.6.1, states that CP applicants should establish a PRA configuration control program to ensure that the CP stage PRA reasonably reflects the preliminary plant design and site characteristics described in the PSAR.

    The NRC Staff's Determination: LME has developed a PRA configuration control program as described in PSAR section 3.1.1.1. The program provides a comprehensive framework for PRA revisions. Although some areas, such as model rollout, peer review, and revision criteria are not fully addressed at the CP stage, the NRC staff determined that the overall configuration control procedure is reasonable to ensure that the PRA will reflect the plant design and site characteristics as they evolve. Therefore, the applicant has met the regulatory position in RG 1.253, section A.6.1.

15. RG 1.253, section A.6.2, states that section 5 of the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, provides an acceptable approach for establishing a PRA configuration control program.

The NRC Staff's Determination: The NRC staff's audit confirmed that the LMGS PRA configuration control program was developed in accordance with section 5 of the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 and therefore, the NRC staff determined that LME has met the regulatory position outlined in RG 1.253, section A.6.2.

16. RG 1.253, section A.6.3, states that the PRA configuration control program may be implemented either as a stand-alone program or included within the quality assurance program (QAP) required by 10 CFR 50.34(a)(7).

    The NRC Staff's Determination: As described in PSAR section 3.1.1.1, the LMGS PRA documentation and software are maintained under a stand-alone LME PRA configuration control process that has an objective of meeting all the supporting requirements within section 5, "PRA Configuration Control," of ASME/ANS RA-S-1.4-2021. The NRC staff's audit verified that the quality control program includes the following elements.

    - Use of qualified personnel for the analyses.
    - Procedures to ensure the control of documentation, including revisions, and provide for independent review, verification, or checking of calculations and information used in the analyses.
    - Documentation and maintenance of records, including archival documentation.
    - Procedures to ensure that appropriate attention and corrective actions are taken if assumptions, analyses, or information used previously are changed or determined to be in error.

    The NRC staff determined that the LME PRA configuration control program is consistent with the discussion provided in NUREG-0800, section 19.0, item 9, and has satisfied the regulatory position in RG 1.253, section A.6.3.

17. RG 1.253, section A.7.1, states that NEI 21-07 provides an acceptable approach and format for presenting CP stage PRA information in the safety analysis report. The PSAR should include a summary justification of the acceptability of the CP stage PRA. This justification should summarize why, commensurate with the facility design maturity, the scope, level of detail, PRA elements, and plant representation are sufficient to implement the LMP methodology for the CP application. Specifically, the summary justification should describe, but is not limited to, the following:

    - How the CP stage PRA was developed in accordance with appendix A of RG 1.253, RG 1.247, and non-LWR PRA standard ASME/ANS RA-S-1.4-2021;
    - How the PRA configuration control program ensures that the CP stage PRA represents the as-designed, as-to-be-built, and as-to-be-operated facility described in the CP application;
    - How the PRA configuration control program will ensure that the PRA for the OL application will represent the as-built, as-to-be-operated facility, including all radiological sources, all hazards, and all POSs, and meet all applicable NRC staff

positions in RG 1.247, and technical elements of the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, not addressed in the CP stage PRA;

- How the applicant's self-assessment of the CP stage PRA was performed consistently with NRC staff positions in section A.8 of RG 1.253.

The NRC Staff's Determination: As described in PSAR section 3.1.1.1, the LMGS PRA was developed following the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, which the NRC staff confirmed in its audit, was documented in Document ID number 006939, "Probabilistic Risk Assessment: Technical Adequacy Summary for the Preliminary Safety Analysis Report," Revision 1. The technical adequacy of the PRA was justified by a PRA self-assessment against the non-LWR PRA standard, as discussed in PSAR section 3.1.1.1.

18. RG 1.253, section A.7.2, states that NRC staff position C.4.1 in RG 1.247, provides an acceptable approach for developing and preserving PRA archival information. PRA documentation providing the detailed justification supporting PRA acceptability should be maintained in archival documentation and, at minimum, include the items described in NRC staff position C.4.1 of RG 1.247.

    The NRC Staff's Determination: As confirmed during the NRC staff audit, while PRA documentation is not fully complete at the CP stage, the justification for PRA acceptability and the items listed in NRC staff position C.4.1 of RG 1.247, are maintained in archival documentation. Therefore, the NRC staff determined that LME has satisfied the regulatory position in section A.7.2 of RG 1.253.

19. RG 1.253, section A.7.3, states that PRA archival information may be controlled through a stand-alone program or the QAP required by 10 CFR 50.34(a)(7).

    The NRC Staff's Determination: As stated in PSAR section 3.1.1.1, the LMGS PRA documentation and software are maintained under a PRA configuration control process that was developed following section 5 of the non-LWR PRA standard ASME/ANS RA-S-1.4-2021. The NRC staff's audit verified that the quality control program includes documentation and maintenance of records, including archival and that the LME PRA configuration control program is consistent with the discussion provided in NUREG-0800, section 19.0, item 9. Accordingly, the NRC staff determined that LME has satisfied the regulatory position in section A.7.3 of RG 1.253.

20. RG 1.253, section A.8.1, states that the guidance in Division of Advanced Reactors and Non-Power Production and Utilization Facilities (DANU) Interim Staff Guidance (ISG) DANU-ISG-2022-05, section 11.1.1, provides an acceptable approach for describing key management responsibilities for PRA development.

    The NRC Staff's Determination: The NRC staff confirmed through audit that the key management responsibilities, as described in PSAR chapter 11, "Conduct of Operations," PSAR section 8.1, and in the LMGS PRA configuration control program (PSAR section 3.1.1.1 describes the LMGS PRA configuration control program), are reasonable. Therefore, the NRC staff determined that LME addressed the regulatory position in section A.8.1 of RG 1.253.

21. RG 1.253, section A.8.2, states that the guidance in DANU-ISG-2022-05, section 11.1.1.1, provides an acceptable approach for describing ability of the CP applicant's technical staff to develop the PRA.

    The NRC Staff's Determination: PSAR section 3.1.1.1 describes the LMGS PRA configuration control program. The NRC staff confirmed through an audit of Document ID number 009473, "Probabilistic Risk Assessment: Configuration Control Program," Revision 2, QAP-3.2, "Technical Analysis Procedure," Revision 6, and QAP-3.20, "X energy Probabilistic Risk Assessment Quality Assurance Procedure," Revision 1, that the responsibilities of the PRA technical staff were documented and that their experience in risk analysis is sufficient to perform PRA. Therefore, the NRC staff determined that the regulatory position A.8.2 was addressed.

22. RG 1.253, section A.8.3, states that the CP applicant should conduct a self-assessment to demonstrate that all PRA logic models, screening analyses, and risk-informed supplemental analyses have been developed and applied in a technically acceptable manner, including the appropriateness of assumptions and approximations.

    a. To support this objective, the self-assessment should provide a basis for asserting that the CP stage PRA is acceptable for implementing the LMP methodology leading up to the CP application submission. The self-assessment should include a review of:

        i. The comprehensive and systematic search used to identify radiological sources, POSs, and hazards;

        ii. The PRA logic models (including scope, level of detail, and elements), screening analyses, risk-informed supplemental evaluations, and credit for DBHLs;

        iii. The CP applicant's PRA configuration control program used to ensure that the CP stage PRA logic models, screening analyses, and risk-informed supplemental analyses represent the as-designed, as-to-be-built, and as-to-be-operated plant.

    The NRC Staff's Evaluation: The LMGS PRA self-assessment, described in PSAR section 3.1.1.1, addressed these items. However, the NRC staff observed several areas for enhancement, as discussed in the LMGS PRA self-assessment subsection above in this SE section 3.1.1.1. Despite areas for enhancement, since a self-assessment was performed and documented that addresses the items listed above, the NRC staff determined that LME sufficiently addressed the regulatory position of RG 1.253, section A.8.3.

3.1.1.2 Conclusion

The NRC staff reviewed PSAR section 3.1.1 and conducted an audit that included the PRA model and its supporting documentation and PRA-related information. The NRC staff concludes that the LMGS PRA follows the guidance in RG 1.247, RG 1.253, and the applicable supporting requirements and HLRs of the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, with one exception as discussed above in PRA element 5 of section 3.1.1.1.

Based on the audit, the NRC staff determined that LME and its PRA team responsible for the LMGS PRA development, have demonstrated general competency in developing the PRA, including the detailed treatment of individual PRA elements. Based on the NRC staff's review of the PSAR as confirmed through the audit, the NRC staff concludes that the processes used to develop the LMGS PRA are logical and structured. The NRC staff also notes that, through LME's self-assessment, LME identified and documented deviations from the non-LWR PRA standard, along with corresponding resolutions, which should be addressed in updates for the OL application stage.

The NRC staff's review determined that the PRA information presented in the PSAR:

1. Aligns with the guidance in RG 1.253, appendix A, and NEI 21-07;
2. Provides a summary of the PRA development, including its scope and level of detail;
3. Describes how PRA results have informed plant design and the safety analysis;
4. Identifies a process for updating and finalizing the PRA;
5. Supports the safety conclusions documented in chapter 4 of the NRC staff's SE;
6. Derives insights to support risk-informed licensing programs; and
7. Serves as a foundation for LMP implementation.

The NRC staff concludes that the PRA information described in the PSAR follows the guidance documents, referenced above, and that the scope, level of detail, and quality of the LMGS PRA are reasonable for the CP application. This conclusion is based on the understanding that the LMGS PRA is preliminary to support the CP application, and that the applicant is not requesting finality at this stage.

In accordance with RG 1.247 and RG 1.253, the NRC staff expects LME to perform the following in support of its OL application, should the applicant apply for an OL:

1. Reassess the IEs to ensure completeness and to include human-induced initiators.
2. Update and expand the PRA scope to incorporate all applicable radiological sources, all internal and external hazards that may affect plant operations, and all POSs relevant to risk evaluations.
3. Update the level of detail in the PRA to accurately reflect plant behavior and operations expected at the OL application stage.
4. Reevaluate assumptions to ensure their continued validity or appropriately address them in the PRA.
5. Update the PRA to reflect the as-designed, as-built, and as-to-be-operated plant configuration.
6. Ensure that the PRA models, quantification, and documentation follow the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 and the regulatory positions in RG 1.247.

7. Reassess sensitivity studies to ensure relevance and update PRA insights accordingly.

8. Update the PRA notebooks and associated supporting documentation to maintain technical accuracy and consistency.

9. Define criteria and triggers for model updates and implement all activities defined in the LMGS PRA configuration control procedure.

10. Conduct a PRA peer review, address all identified findings and observations, and develop a resolution plan for any long-term issues.

11. Update all applications that rely on PRA results and insights to reflect the revised PRA.

In summary, based on the NRC staff's review, audit, and evaluation described above, the NRC staff determined that the LMGS PRA is reasonably developed and maintained, and provides sufficient support for the issuance of LMGS CP.

3.1.2 Design-Basis Accident Analytical Methods

PSAR section 3.1.2 summarizes the evaluation models (EMs) developed in accordance with RG 1.203 that are used to evaluate DBAs for the LMGS design. The PSAR references Topical Report (TR) 007834, "Xe-100 Licensing Topical Report Transient and Safety Analysis Methodology," Revision 2 (ML25206A049) (hereafter referred to as the Transient and Safety Analysis Methodology (TSAM) TR), TR 008585, "Xe-100 Licensing Topical Report GOTHIC and Flownex Analysis Codes Qualification," Revision 3 (ML25204A205) (hereafter referred to as the GFQ TR), and TR 000632, "Xe-100 Licensing Topical Report Mechanistic Source Term Approach," Revision 3 (ML25195A103) (hereafter referred to as the MST TR) for details regarding the DBA analytical methods and the EMs.

The TSAM TR describes the overall EM used in DBA analyses, encompassing both the thermodynamic analysis and the MST analysis (i.e., radionuclide transport and release). The GFQ TR provides the thermodynamic analysis models developed using the GOTHIC and Flownex codes, while the MST TR presents the mechanistic source term methodology developed using the XSTERM code. Section 3.1.3 of this SE discusses the thermodynamic analysis approach and the qualification of GOTHIC and Flownex codes, and section 3.1.4 of this SE discusses the evaluation of the XSTERM code and the MST methodology. Section 3.1.5 of this SE discusses the atmospheric dispersion and dose calculation methodology and section 3.1.6 of this SE discusses the core design and analysis methodology, which completes the codes and methods that provide input to the DBA EM.

The applicable regulatory requirements for evaluation of DBA analytical methods are as follows:

- 10 CFR 50.34(a)(1)(ii)(D);
- 10 CFR 50.34(a)(4);
- 10 CFR 50.34(a)(3);
- 10 CFR 50.34(a)(8);

- 10 CFR 50.43(e)(1)(iii);
- 10 CFR 50.35(a)(2); and
- 10 CFR 50.35(a)(4).

The NRC staff notes that while 10 CFR 50.43(e) is not applicable to a CP application, the completion of the research and development (R&D) activities discussed under 10 CFR 50.34(a)(8) may be, in part, necessary for the required performance demonstrations of the critical safety systems in compliance with 10 CFR 50.43(e), at the OL application stage.

The guidance applicable to the evaluation of DBA analytical methods is as follows:

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

The principal design criteria (PDCs) applicable to the evaluation of DBA analytical methods are as follows:

- PDC 10, “Reactor design”
- PDC-RFDC [Required Function Design Criterion] 11, “Reactor inherent protection”
- PDC-RFDC 16, “Functional containment design”
- Xe-100 PDC 19, “Control room”
- PDC 20, “Protection system functions”
- PDC-RFDC 26, “Reactivity control systems”
- PDC 28, “Reactivity limits”
- PDC-RFDC 30, “Integrity of reactor helium pressure boundary”
- PDC-RFDC 34, “Residual heat removal”

3.1.2.1 Technical Evaluation

The NRC staff evaluated the implementation of the TSAM TR to determine compliance with the above regulations and to assess its suitability for performing the preliminary safety analysis of DBAs for the Xe-100 reactor design (ML25062A069, ML25206A049). The NRC staff concluded that the TSAM TR complied with the identified regulations and PDCs and approved its use, subject to three limitations and fourteen conditions. The evaluation presented here is limited to verifying that the limitations and conditions (L&Cs) associated with the TSAM TR are addressed adequately in the LMGS PSAR.

PSAR table 1.4-11 provides the applicant’s plan for implementing the evaluation model development and assessment process (EMDAP) steps in RG 1.203 and to address the TSAM TR L&Cs. The NRC staff finds that TSAM is used in the PSAR preliminary analyses consistent with the limitations in the TSAM TR because:

- The TSAM EM is applied to the preliminary safety analysis of the LMGS reactor which is based on the Xe-100 design.
- The H2TS scaling methodology is used only for general scaling considerations.
- The TSAM EM is only used to generate thermal-fluid inputs for the radiological consequence assessment using the XSTERM code.

The TSAM TR conditions 3, 4, 5, 7, 8, 9, 10, 13 and 14 relate to completion of activities associated with the EMDAP steps 1 through 20 and the need for identification of the remaining scope of work as a part of a relevant R&D program consistent with 10 CFR 50.34(a)(8). However, as summarized in PSAR table 1.4-11 and based on the criteria presented in PSAR chapter 13, "Research and Development Program," the applicant states that these conditions do not meet the definition of an R&D program under 10 CFR 50.34(a)(8). The NRC staff evaluated this position in SE section 13.1.1 and noted disagreement with the applicant's interpretation of the applicability of 10 CFR 50.34(a)(8) for identifying R&D items.

The NRC staff's review of PSAR table 1.4-11 finds that the applicant identifies the outstanding scope of work associated with all relevant TSAM conditions - including gaps in PIRT, testing needs, model development, scaling and distortion analysis, validation, applicability evaluation, and uncertainty quantifications - and commits to completing these activities at the OL application stage. These commitments provide reasonable assurance that safety questions associated with SSC adequacy and integral system performance for DBA mitigation will be addressed during the OL application review. To gain additional insight into the applicant's plans for completing the EMDAP steps beyond the summary provided in the PSAR, the NRC staff also audited the LME document titled, "Xe-100 Regulatory Guide 1.203 Project Execution Plan," Document ID number 010945. While the NRC staff does not make determination regarding the applicant's exclusion of TSAM conditions 3, 4, 5, 6, 7, 8, 9, 10, 13 and 14 from the R&D program under 10 CFR 50.34(a)(8), the NRC staff finds that the scope of planned activities described in PSAR table 1.4-11, can reasonably be provided in the final safety analysis report (FSAR) in accordance with 10 CFR 50.35(a)(2). Furthermore, consistent with 10 CFR 50.35(a)(4), the NRC staff has reasonable assurance that, if the activities identified in PSAR table 1.4-11 are completed as described, the safety questions associated with the DBA EM will be satisfactorily resolved and the proposed facility can be constructed and operated without undue risk to public health and safety.

Condition 1 of the TSAM TR is satisfied because the PDCs evaluated in the TSAM TR remain unchanged in the PSAR. Condition 2 is satisfied because all DBA events analyzed in the PSAR are within the scope of the TSAM EM. Conditions 11 and 12 are satisfied because PSAR table 1.4-11, explains the basis for FSC and provides justification for applying them to all the DBAs.

Based on this review, the NRC staff concludes that all TSAM L&Cs are addressed adequately in the PSAR for the CP application.

3.1.2.2 Conclusion

The NRC staff evaluated the TSAM DBA EM for the Xe-100, as described in TR 007834, and determined that LME's use of the TR for LMGS complies with the applicable regulations. Based

on the technical evaluation summarized above, the NRC staff concludes that all the L&Cs on the TSAM TR are addressed adequately in the PSAR for the CP application.

Furthermore, the NRC staff finds the use of the TSAM EM acceptable for: (1) informing radiological consequence consistent with requirements of 10 CFR 50.34(a)(1)(ii)(D), and (2) supporting the preliminary safety analysis of the design and performance of LMGS design SSCs in accordance with 10 CFR 50.34(a)(4), because LME references approved methodologies; and LME addressed the L&Cs associated with the referenced methodologies.

3.1.3 Thermodynamic Analysis

PSAR section 3.1.3 describes the applicant's approach for evaluating the plant's thermodynamic behavior during LBEs. In developing these analyses, the PSAR incorporates the GFQ TR (TR 008585), which describes the GOTHIC and Flownex codes and the associated input models developed in accordance with the RG 1.203 EMDAP.

The applicable regulatory requirements for the evaluation of GOTHIC and Flownex thermodynamic analyses model are as follows:

- 10 CFR 50.34(a)(4); and
- 10 CFR 50.35(a)(2).

The guidance applicable to the evaluation of GOTHIC and Flownex thermodynamic analyses is as follows:

- RG 1.203.

3.1.3.1 Technical Evaluation

The NRC staff's evaluation of the GFQ TR approved the use of TR 008585 for the preliminary safety analysis of DBAs consistent with 10 CFR 50.34 (a)(4) for the Xe-100 design, subject to one limitation (ML25262A060,). This limitation requires that the review of input parameters for the Flownex and GOTHIC input models be performed as part of the review of the PSAR. Accordingly, the evaluation presented here is focused on verifying that the GFQ TR limitation is addressed adequately in the PSAR for the LMGS reactor design, confirming that the Flownex and GOTHIC models used in the PSAR analyses remain consistent with those approved in the GFQ TR, and evaluating any new modeling elements introduced in the PSAR that were not previously reviewed in the GFQ TR.

The thermodynamic analysis methodology follows the EMDAP in RG 1.203 and includes two system analysis codes - Flownex and GOTHIC - to evaluate short-term and long-term responses of the plant under postulated transient conditions. In the short-term phase, the applicant uses the Flownex thermal-hydraulic code to model transient behavior over periods ranging from several minutes to multiple hours, during which plant parameters are still evolving due to control and shutdown rod motion, valve realignments, circulator behavior, and changes in SSC status. This phase represents conditions when active systems may continue to influence the plant response. If the short-term analysis demonstrates that forced cooling is maintained at the end of the transient, the applicant does not perform an additional long-term evaluation.

If, however, the Flownex end state reflects transition to a non-forced-circulation condition, the long-term passive-cooling phase is evaluated using the GOTHIC code. In this phase, the applicant models the plant's behavior for up to 30 days, representing bounding LBE conditions. The PSAR explains that in the long-term phase, the plant parameters are not significantly changing, and the core heat is removed passively.

Importantly, the PSAR clarifies that both the Flownex short-term and GOTHIC long-term analyses begin at time zero. There is no handoff or data transfer from the short-term simulation to the long-term simulation. Instead, each model independently initializes from conservative bounding conditions for its respective analytical purpose. The PSAR further clarifies that the total dose is calculated by adding the dose from both Flownex and GOTHIC. This double counting of the dose for the short-term period adds additional conservatism to the results.

The NRC staff reviewed the input parameters for the GOTHIC and Flownex models to support its review of the short-term and long-term response of the plant under postulated transient conditions described in the PSAR. During the audit, the NRC staff reviewed the following Flownex and GOTHIC input model calculation notebooks and safety analysis:

- Document ID number 003221, "Preliminary Safety Analysis Flownex Base Model Report," Revision 3.
- Document ID number 005504, "Preliminary Xe-100 GOTHIC8.4a (QA) Reactor System Model Report," Revision 3.
- Document ID number 00213, "Preliminary Flownex Safety Analysis of a Steam Generator Tube Rupture," Revision 4.
- Document ID number 004276, "Large depressurization Analysis-Licensing Basis Events," Revision 3.
- Document ID number 002136, "Small and Medium Depressurization Licensing Basis Event Safety Analysis Calculation," Revision 5.

The NRC staff audited the applicant's base model calculation notebooks (Document ID numbers 003221 and 005504) to review the geometric, model-correlation, and initial and boundary condition input parameters used in the PSAR analyses. The notebooks document the sources of the selected input parameters and identify key assumptions applied where data were unavailable. Based on this audit, the NRC staff confirmed that the Flownex and GOTHIC models used in the PSAR are consistent with the models approved in the GFQ TR.

In addition, the NRC staff also evaluated several modeling aspects not fully addressed in the GFQ TR, including treatment of material properties, modeling of steam ingress phenomena for the SGTR event, air ingress phenomena in depressurized loss of forced cooling (DLOFC) events, and the modeling of RCCS. The NRC staff's evaluation of these modeling approaches is discussed below.

The NRC staff reviewed PSAR section 3.1.3, which clarifies that the material properties used in the GOTHIC and Flownex models account for temperature-dependence and irradiation-induced effects. The PSAR incorporates uncertainties in key material properties through the PKPIRT process. Specifically, graphite thermal conductivity inputs are based on graphite material

qualification testing and vendor-supplied specifications and accounted for dependencies on irradiation temperature history, fast neutron fluence, and transient temperature. Metallic material properties are taken from ASME Section II specifications and vendor data. The NRC staff determined that this approach is acceptable for preliminary analysis; however, for the OL application, material properties are expected to be based on applicable graphite and metallic qualification testing.

The NRC staff reviewed PSAR section 3.1.3, which explains that an air-cooled RCCS model is used in the PSAR preliminary analyses rather than the water-cooled RCCS used in the design. The PSAR states that this approach results in a conservative analysis because the water-cooled RCCS is expected to provide better heat transfer due to the large heat transfer area and higher convective heat transfer performance. The PSAR states that OL application analyses will be based on the water-cooled RCCS. The NRC staff finds this RCCS modeling approach acceptable for the preliminary analyses, as it is expected to provide conservative results for the DBAs evaluated in the PSAR. Additional evaluation of RCCS design, modeling approach, and remaining scope for OL application is provided in SE sections 6.4.11.

The NRC staff reviewed PSAR section 3.1.3, which discusses the applicant's modeling approach for steam ingress phenomena in SGTR events, including two-phase flow and gas mixture transport, graphite oxidation, and critical flow. PSAR section 3.1.3 confirms that the validation and applicability assessments of GOTHIC and Flownex for the steam ingress phenomena will be performed for OL application, consistent with the EMDAP framework outlined in GFQ TR. The NRC staff finds this identification of remaining scope of work and commitment to completing it at the OL application stage reasonable and in accordance with 10 CFR 50.35(a)(2).

The NRC staff reviewed PSAR section 3.6.11, which discusses the applicant's air ingress analysis in the DLOFC events. PSAR section 3.6.11 states that the RB is modeled conservatively as a fixed boundary at atmospheric pressure with 100 percent air, consistent with the GFQ TR. This assumption maximizes potential air ingress by not crediting helium mixing and transportation within the RB. The PSAR commits to evaluating break size, location, orientation and multi-dimensional flow effects on air ingress and graphite oxidation at the OL application stage. The NRC staff finds the applicant's air ingress modeling approach for the DLOFC event in PSAR analyses acceptable because it provides a bounding representation of the RB air available for ingress at the break. Additionally, the identification of the remaining scope of work and commitment to complete it at the OL application stage is reasonable and in accordance with the requirements of 10 CFR 50.35(a)(2).

The NRC staff performed confirmatory MELCOR calculations, as discussed in SE sections 3.1.4 and 3.6, for key DBA scenarios to evaluate GOTHIC and Flownex predictions presented in the PSAR. These calculations assessed sensitivity to important parameters such as thermo-physical properties, reactivity coefficients and operating conditions. As discussed in section 3.6 of this SE, the results show reasonably good overall agreement between the MELCOR model and the PSAR analyses.

3.1.3.2 Conclusion

The NRC staff finds the use of Flownex and GOTHIC acceptable for evaluating LBEs and performing the preliminary safety analysis of the design and performance of LMGS SSCs in accordance with 10 CFR 50.34(a)(4). This determination is based on:

(1) the NRC staff's prior approval of these codes for preliminary DBA analysis for the Xe-100 design, which forms the basis for the LMGS reactor design;

(2) the NRC staff's conclusion that the GFQ TR limitation is adequately addressed in the PSAR;

(3) the PSAR's satisfactory treatment of several modeling aspects not fully covered in the GFQ TR, along with the applicant's commitment to complete the remaining scope of work for the OL application consistent with 10 CFR 50.35(a)(2); and

(4) the fact that the EMDAP applied to the Xe-100 DBA analyses under RG 1.203 is more rigorous than the process required for LBE analysis.

3.1.4 Mechanistic Source Term Methodology

PSAR section 3.1.4, describes the set of MST analysis models used to quantify the Xe-100 source terms for use in the evaluation of dose consequences from LBEs, including models for DBAs and non-DBA LBEs, for the LMGS.

The following TR is incorporated by reference (IBR) into PSAR section 3.1.4:

- Xe-100 Licensing Topical Report, "Mechanistic Source Term Approach," Revision 3 (ML25073A097)

The regulatory requirements for the evaluation of the MST are as follows:

- 10 CFR 50.34(a)(1) ;and
- 10 CFR 50.34(a)(4).

The applicable guidance for the evaluation of MST is as follows:

- RG 1.233, Revision 0, which endorses NEI 18-04, Revision 1, with clarifications;
- RG 1.253, Revision 0, which endorses NEI 21-07, Revision 1, with clarifications and additions; and
- RG 1.247, Revision 0, which endorses the non-LWR PRA standard ASME/ANS RA-S-1.4-2021.

Section 2 of RG 1.183, Revision 1 (ML23082A305) provides useful information regarding the development of non-scenario-specific source terms for major accidents or maximum hypothetical accidents. Such source terms are used in consequence analyses to show compliance with the 10 CFR 50.34(a)(1)(ii)(D) safety analysis requirements, as well as the non-seismic site criteria postulated accident analysis requirement in 10 CFR 100.21(c)(2), which references the safety analysis dose criteria in 10 CFR 50.34(a)(1).

3.1.4.1 Technical Evaluation

Radiological source term refers to the type, quantity, and timing of the release of radioactive material from a facility to the environment during a postulated event. The composition and

chemical form of the released material are also defined. Traditional LWR source term evaluations assumed the release of radioactive material from its initial location (e.g., reactor fuel) during an accident using deterministic methodologies. For a major accident in an LWR, the released material was traditionally defined as an instantaneous release into the containment, but is defined as a release over time (see RG 1.183, Revision 1) into the containment in more recent LWR evaluations. In addition, credit was given for attenuation of the released material prior to it reaching the environment, consistent with the plant design features (e.g., suppression pool, containment sprays) and physical properties (e.g., decay) of the radioactive material.

The industry guidance in NEI 18-04 proposed a more realistic (mechanistic) approach for determining the radiological source terms for non-LWR ESs through an integrated assessment of the integrity of the reactor fuel and, upon fuel failure, the subsequent release, nature, transport and attenuation of the radioactive material prior to reaching the environment. The proposed approach relies on having sufficient knowledge (e.g., data, analysis, operating experience) regarding the release of the radioactive material from its initial location during an accident to accurately model and to calculate the release as a function of time (or as a function of some other parameter, such as fuel temperature), including its composition, chemical form, and uncertainty. Radiological source terms may also be specific to an LBE type, POS, or location of the radionuclides in the proposed approach.

The NRC staff reviewed the technical basis for the MSTs proposed for use in the PSAR. The NRC staff's SE of the Xe-100 MST TR (ML25195A103) concluded that the failure probability model (FPM), solids fission product transport model (SOLM), gaseous fission product transport model (GASM), dust production model (DUSTM), helium pressure boundary model (HPBM), and core corrosion calculations model (CORRM) contained in the XSTERM computer code appear to cover the phenomena needed to predict the MST to support the preliminary analysis and evaluation of the Xe-100 design. However, the NRC staff's SE made no conclusions regarding the acceptability of the models contained in XSTERM because: (1) the XSTERM models are under development, (2) the NRC staff did not complete a detailed technical review of the models, (3) the details regarding key phenomena identification and associated knowledge levels are not provided in the TR, and (4) the models and associated validation plans are preliminary and subject to change. The NRC staff's SE on the MST TR states that a detailed technical review of XSTERM model applicability to the Xe-100 reactor will be performed as part of the review of a licensing application that references this TR. The NRC staff's review of the preliminary XSTERM models in support of the CP application is documented below.

The NRC staff's evaluation of the methodology used by LME to develop MSTs for the PSAR analyses are as described in the following subsections of this SE:

1. Reviewing the acceptability of the models applied for development of MSTs (section 3.1.4.1.1 of this SE)

2. Reviewing the radionuclide inventories and source terms used in LBE evaluations (sections 3.1.4.1.2 through 3.1.4.1.5 of this SE)

3. Reviewing the uncertainties applied to MST evaluations (section 3.1.4.1.6 of this SE)

4. Reviewing the functional containment strategy and the related radionuclide release barriers applied to MST (section 3.1.4.1.7 of this SE)

In addition, the NRC staff audited documentation related to the MST development. The NRC staff audited information on how LME determined pebble inventory, HPB inventory, the application of functional containment for the design, and how the source term was modified to reflect each LBE.

3.1.4.1.1 The NRC Staff's Review of the Xe-100 MST TR Models Applied for Development of the Mechanistic Source Term

PSAR section 3.1.4 summarizes the use of the XSTERM code to calculate the MST for LMGS and states that more detailed information regarding the calculations of the MST can be found in the Xe-100 Licensing TR, "Mechanistic Source Term Approach," Revision 3, which is IBR as reflected in PSAR table 1.4-4.

The NRC staff reviewed the implementation of models described in the Xe-100 MST TR Revision 3, and confirmed the basis of the models used to develop the LBE source terms described in the PSAR, including the methods in the XSTERM calculational code and the code modules that evaluate different source-term-related phenomena. LME describes the XSTERM code operation as follows: Power and neutron fluence and global thermal hydraulic parameters are taken as input to XSTERM. XSTERM then evaluates tri-structural isotropic (TRISO) and pebble temperature distributions with the thermodynamics calculation model (THM), evaluates TRISO failures using the TRISO particle FPM, evaluates radiological releases using the GASM and SOLM models, evaluates carbon and metallic dust generation using the DUSTM, evaluates the distribution of released fission products and the response to events using the HPBM, and evaluates the impact of oxidation for events that involve steam or air contact with fuel using the CORRM.

The XSTERM code is used to perform: (1) steady-state calculations evaluating the radionuclide inventories in the pebbles and HPB over 60 effective full power years (EFPY), and (2) transient calculations using the steady-state inventories as initial conditions for the evaluation of the radiological releases to the environment (source term) caused primarily by the unique temperature profile of the fuel during the LBEs. The steady-state inventory calculations consist of tracking pebbles through the core, evaluating their temperature and burnup histories, evaluating the production; decay; and transmutation of radionuclides in these pebbles, evaluating carbon and metallic dust generation and distribution within the HPB, and evaluating the release of the generated radionuclides to the HPB. The source term calculations consist of modeling the transport of radionuclides within the Xe-100 reactor due to thermodynamic conditions for the specific transient scenarios to the environment.

Table 3.1-1 below provides the mapping of major source-term-related phenomena to the associated XSTERM code models and the equivalent model used in the Xe-100 MST TR. The summary of the general approach and models used by LME to establish steady-state radionuclide inventories is provided in table 3.1-2. The summary of the general approach and models used by LME in the XTERM code to evaluate the resulting source term from transient events is provided in table 3.1-3. These tables provide a guide to relate the different models to relevant phenomena and the role of these phenomena and models in the evaluation of the MST for various analyzed transients.

Table 3.1-1: Mapping of significant ST-related phenomena to XSTERM and Xe-100 MST TR models

Phenomenon	Description	XSTERM Model/ MST TR Model
Initial TRISO defects	Initial, user specified, TRISO manufacturing defects and heavy metal contamination.	XFP/FPM
Neutron flux/fluence and power distribution	Generated externally – input to XSTERM from Very Superior Old Programs (VSOP) code.	-
Fluid flow, heat transfer	Fluid flow and large-length-scale heat transfer input to XSTERM for operating and transient conditions from GOTHIC and Flownex codes.	-
Decay heat	Decay heat correlation.	XTDYN/THM
Heat transfer	Heat transfer by conduction, convection, and thermal radiation used to evaluate core-wide pebble temperature and TRISO distributions. The calculation involves material properties conductivity, density, heat capacity with variation in properties due to overall burnup and/or fast fluence.	XTDYN/THM
Pebble Tracking	Tracking pebble transport through core and tracking their burnup and temperature histories.	XSOL/SOLM
TRISO failures during operation	Evaluation of TRISO failures during operation based on temperature and fluence.	XFP/FPM
Radionuclide inventory	Evaluation of radionuclide production and destruction of dose-relevant radionuclides from fission events, transmutation, and radioactive decay based on changing flux, temperature.	XSOL/SOLM
Tritium generation	Tritium generation by different mechanisms.	XTRIT/TRITM
Radionuclide distribution in pebbles	Evaluation of fission product distribution in pebbles and escaped from pebbles based on fission-product diffusion through different materials as a function of temperature, and flux histories experienced by pebbles.	XSOL/SOLM
Dust generation	Generation of carbon and metallic dust due to friction.	XDUST/DUSTM
Radionuclide distribution in HPB	Evaluation of the distribution of radionuclides in the HPB. This includes the evaluation of fission product sorption on dust and HPB surfaces, lift off; steam off, and wash off, evaluation of retention and filtration (e.g. cryotraps), and release. Radionuclides reaching the HPB are evaluated by XSOL. Requires thermal hydraulic information about pressure, temperature, and flow from Flownex and/or GOTHIC analyses.	XHPB/HPBM

	Radionuclides reaching the HPB are evaluated by XSOL.	
Influence of burnup transients on TRISO failure	Higher burnup leads to more gas inventory in TRISO and can lead to Silicon Carbide (SiC) corrosion. Higher temperatures lead to higher gas pressures and thus higher stresses on SiC, resulting in additional TRISO failures.	XFP/FPM
Oxidation	Oxidation by trace oxygen during operation, oxidation by oxygen during accidents involving breaks, and oxidation by steam for SGTRs that also results in hydrogen generation.	XCORR/CORRM
Reactor building decontamination	- not modeled	-
Atmospheric dispersion and dose	Evaluation of atmospheric dispersion and dose from the different source terms.	XDIS
	Note: XGAS/GASM which evaluates release over birth ratios for gases, does not seem to be directly used. Rather, the models are used to derive diffusivities for XSOL/SOLM.	

Table 3.1-2: General process for the radionuclide inventory calculation in XSTERM

Phenomenon	XSTERM Model/ MST -TR Model
Pebble heavy metal contamination and initial TRISO failures set by user.	XFP/FPM
Evaluation of the initial pebble inventory for 60 EFPY using: • XTDYN/THM for higher resolution temperatures from those evaluated with Flownex, GOTHIC, and/or VSOP. • XSOL/SOLM evaluates transport of pebbles through core, tracks pebble temperature and flux history, and evaluates buildup of radionuclides based on yields, decay chains, and neutron capture from flux obtained from VSOP results. • XCORR/CORRM can be used to evaluate oxidation of graphite from trace oxygen in the system.	XTDYN/THM XSOL/SOLM
XDUST/DUSTM used to evaluate generation of carbon and metallic dust.	XDUST/DUSTM
XHPB/HPBM used to evaluate the distribution of fission products and dust in the HPB.	XHPB/HPBM
XTRIT/TRITM used to evaluate tritium production and transport.	XTRIT/TRITM

Table 3.1-3: General XSTERM process for reactor transient calculations (e.g. DLOFCs, PLOFCs, SGTRs)

Phenomenon	XSTERM Model/ MST TR Model
Start out with the inventory developed from the steady-state calculations and operating conditions.	
XTDYN/THM is used to evaluate temperatures histories for pebbles and TRISO layers at different locations within the core based on TH calc results (Flownex and GOTHIC). Establishes time-dependent temperatures for each pebble group at each location.	XTDYN/THM
XFP/FPM calculates additional TRISO failures due to transient conditions.	XFP/FPM
XSOL/SOLM Calculates the release fraction from pebble to HPB. SOLM calculates additional releases based on the temperatures evaluated by XTDYN/THM, additional pebble failure calculated by XFP/FPM, and the evaluation of local temperature-dependent diffusivities.	XSOL/SOLM
XCORR/CORRM is used to evaluate the additional enhancement to release due to oxygen for DLOFC, steam for SGTR. Evaluates flammable gas (hydrogen and carbon monoxide) generation for SGTR events.	XCORR/CORRM
XHPB/HPBM calculation of leakage of fission products outside of HPB based on either assumed leakage rate (specified) or Thermohydraulic code (GOTHIC or Flownex) for breaks. XHPB/HPBM also calculates releases from HPB surfaces by liftoff and calculates steamoff and washoff during SGTR events. Thermohydraulic calculations also account for breathing of system which assumes a certain amount of cooldown and reverse flow for maintaining HPB inventory.	XHPB/HPBM
Retention in the RB is not credited.	-
XDIS is used to calculate dose.	XDIS

The NRC staff compared the physical models listed in the Xe-100 MST TR to other known models relevant to the listed source-term-related phenomena. This comparison provided information about the appropriateness of the models, but not reasonability of overall results or whether the implementation matches the model description.

To assess the reasonability of the results, the NRC staff conducted independent source term calculations for certain key accident scenarios using both the MELCOR and SCALE codes, both of which are widely used to develop source terms for nuclear reactor accident analyses. This involved independent evaluation of the pebble and HPB radionuclide inventories, the spatial power distribution, reactivity coefficients, and decay power using the SCALE code. Many of the physical models used in MELCOR and SCALE were developed and updated for Next Generation Nuclear Plant (NGNP) and the 400 MWth Pebble Bed Modular Reactor (PBMR-400) benchmark. Some of this previous work is described in the following documents:

- "PBMR Coupled Neutronics/Thermal-hydraulics Transient Benchmark - The PBMR-400 Core Design," NEA/NSC/DOC(2013)10, July 2013.
- Bostelmann, F., et al., "SCALE Capabilities for High Temperature Gas-Cooled Reactor Analysis," Annals of Nuclear Energy, Volume 147, 107673, November 2020.
- M. Young, et al., MELCOR Fission Product Release Model for HTGRs, SAND2010-0800C
- K. Wagner, B. Beeny, and D. Luxat, MELCOR Accident Progression and Source Term Demonstration Calculations for HTGR, SAND2022-2750, R1: 2022. (R2 of this document was released in 2024.)

A discussion of the results of the Xe-100 source term confirmatory calculations can be found in SE section 3.1.4.1.5.

The XSTERM code uses power distribution and reactivity coefficients evaluated with the VSOP reactor physics codes, and flow and temperatures evaluated with the GOTHIC and Flownex thermal hydraulic codes to evaluate local temperatures used to evaluate fuel failures and fission product release rates, using diffusion coefficients. The NRC staff's evaluation of the thermodynamic analysis in the PSAR is contained in SE section 3.1.3. The predicted radiological releases are highly sensitive to these inputs and uncertainties in these inputs propagate to uncertainties in predicted radiological releases. The NRC staff's assessment of the uncertainty of the source term modeling is contained in SE section 3.1.4.3.6.

3.1.4.1.1.1 Thermodynamics Calculation Model

The NRC staff reviewed the XSTERM THM which evaluates the temperature distribution in fuel pebbles, TRISO-coated particles, and all core components such as reflectors, barrel, and reactor pressure vessel (RPV) during normal and LBE conditions. The THM evaluated temperature distributions which are used by fuel failure and fission product release models (e.g., temperature-dependent diffusion coefficients). The THM model is also used in situations where the use of detailed thermal hydraulic models is not practical.

THM evaluates temperature based on time-dependent temperature and flow distributions. The global power distributions along with reactivity coefficients are evaluated using the VSOP reactor physics code. To evaluate local temperatures, THM uses as input thermal and flow information evaluated by the Flownex and GOTHIC codes using the VSOP-evaluated power distribution and German standard decay power curve (DIN 25485), which is a standard correlation used for these phenomena. Flownex is used for short-term thermal hydraulic analyses whereas GOTHIC is used for longer term thermal hydraulic evaluations.

Section 5.1.3 of the Xe-100 MST TR discusses the THM module and its calculations, more details and equations are provided in TR appendix B. For steady-state or near steady-state conditions, the temperature distributions with solids depend solely on the conductivity. Therefore, the NRC staff's review prioritized thermal conductivities over heat capacities/thermal diffusivities when reviewing the steady-state source term calculations. Therefore, the NRC staff's review prioritized thermal conductivities over heat capacities/thermal diffusivities when reviewing the steady-state source term calculations. The NRC staff notes that the documentation provided in appendix B of the Xe-100 MST TR describes, among other

parameters, the evaluation of pebble residence times within the core, fuel radionuclide buildup, group neutron flux, evaluations of fast fluence, particle temperature, fuel-element temperature, and coolant temperature, some of which influence material thermal properties.

The NRC staff notes that LME obtained the pebble conductivity model from **[[**

]]. TR table B-7 states that the Mitchell-Haag model is used for the heat conduction coefficient of reflector graphite and TR appendix B.2.2.4.4.1, describes the derivation of the Mitchell-Haag model for heat conduction coefficient of reflector graphite. Haag notes that the model is "based upon data of numerous graphite grades that were irradiated in the Petten High Flux Reactor from 1970 till 1990."

Based on the above evaluation, the NRC staff determined that the described physical models used to evaluate TRISO, pebble, and system temperature distributions in THM originated from, and are consistent with, codes and validated models previously used to evaluate these temperature distributions and therefore are appropriate for describing the time-dependent temperature effects needed for MST analyses.

3.1.4.1.1.2 Failure Probability Model

The NRC staff reviewed the XSTERM FPM which calculates particle failure probabilities during normal operations and LBE conditions. Particle failure probabilities are calculated for the failure mechanisms including SiC pressure vessel failure due to internal gas pressure in the particle, kernel migration (amoeba), fission product corrosion, thermal decomposition, and inner pyrolytic carbon (IPyC) cracking.

The assumed manufacturing defects and heavy metal contamination in the pebble matrix are treated as fuel failures since the fission products resulting from defects, contamination, and failed fuel are more readily released because they do not have the TRISO layer barriers to release that intact TRISO particles have. PSAR section 3.3 specifies the allowable initial fuel failure fraction of 1.1×10^{-4}, which is the maximum value acceptable for the Xe-100 fuel specification.

The NRC staff notes that section 5.1.1 and appendix A of the MST TR, Revision 3, describes the FPM. The MST TR describes the use of the following methodology to calculate failure of TRISO particles: The XSTERM TRISO failure model is based on **[[**
]], SiC strength of particles described by a Weibull-distribution is challenged by stresses applied by fission gases accounting for the initial layer compression, weakening of SiC by irradiation, and thinning of layers by fission product corrosion and thermal decomposition. Although also modeled, the kernel migration (amoeba) effect and carbon monoxide (CO) corrosion is considered to be negligible for the TRISO-X fuel due to the expected 0 net oxygen generation from fissions occurring in Uranium Oxycarbide (UCO) fuel, as supported by **[[**
]].

During the audit, the NRC staff confirmed that XSTERM uses the same coefficients as **[[**

]]. The XSTERM SiC strength for Xe-100 after considering radiation effects in the model, is greater than that 480 MPa (irradiated) SiC strength in the US model described in IAEA TECDOC-978, table A-8.

During the audit, the NRC staff confirmed the tensile strength in SiC, s_t, is calculated by LME to be less than 0 during operation indicating that SiC experiences compressive stresses and thus no failures are predicted. **[[**

]] the LME approach for the evaluation of layer stresses as described in the Xe-100 MST TR seems to differ from the original approach in (Verfondern and Nabielek, Mathematical Basis of the PANAMA 1 code for modeling pressure vessel failure of TRISO-coated particles under accident conditions, HTA-IB-03/90, 1990), which may result in a deviation from the original model whose strength curve may be valid only in conjunction with its stress model. The NRC staff will review the TRISO failure model in more detail during a future OL application review that compares behavior to the available data considering the statistical variability. The SiC corrosion model is described in Xe-100 MST TR, appendix A, section 2.3.1.2. **[[**

].

Based on the models described in the Xe-100 MST TR, as confirmed by the information reviewed during the audit, the NRC staff determined that the models used to determine fuel failures appear reasonable however, validation against data is needed to ascertain how well the model predicts TRISO fuel failures encompassing the operational envelope in terms of temperature, burnup, and fast fluence in US UCO fuel including for postulated accident conditions. During a future OL application review, the NRC staff will review XSTERM models used to determine fuel failures to confirm that the predicted lack of failures during operation matches observations for this type of fuel. During a future OL application review, the NRC staff will also confirm that the strength values correspond to those previously used for SiC in US TRISO and that the approach appropriately accounts for uncertainty (e.g. 95th percentile) in TRISO failure observations.

3.1.4.1.1.3 Solids Fission Product Transport Model

The NRC staff reviewed XSTERM SOLM which includes a radionuclide diffusion model used for calculating production, decay, transmutation, transport, and leakage of gaseous and solid fission products from fuel pebbles into the HPB under steady-state and LBE conditions. In addition, the model calculates solid fission product release in pebbles transporting through the core. SOLM uses the TRISO and pebble temperature distributions evaluated by the THM thermodynamics model to determine the fission product diffusivities in the different TRISO and pebble regions. TRISO-coated particle failure probabilities are integrated from the FPM particle failure probabilities model. The HPBM tracks the inventories and the distribution of fission products that SOLM calculates to be released from the pebbles to the HPB.

The Xe-100 MST TR Revision 3, discusses SOLM in section 5.1.2 and in appendix C. Section C.2.1 of the TR lists the discretization for TRISO particles, fuel pebbles, and fuel compacts used for the evaluation of fission product release. The NRC staff finds the discretization to be similar to that for heat transfer in THM. The diffusion model in section C.2.2 includes removal by neutron capture and radioactive decay and sources from fission, activation, and decay. Fission product release occurs not only by diffusion through barriers but more rapidly through failed particles. When the SiC layer is failed, the entire TRISO particle is considered to have failed. Since transport of some fission products behave differently depending on whether just the SiC layer fails or the other PyC layers, not distinguishing between SiC and full TRISO failures is conservative because the additional retention provided by the intact PyC layers is not considered. The fission-product diffusivities used in the SOLM are not explicitly listed in the Xe-100 MST TR although the following report is referenced for diffusivities: **[**

]]. During the audit, the NRC staff reviewed the list of SOLM diffusion coefficients used in the PSAR analyses for comparison to NRC confirmatory calculations. These included additional diffusivities not included in the Xe-100 MST TR references and also included some diffusivities back-calculated from GASM results. This approach is also referenced in the Xe-100 MST TR, appendix H where the Xe-100 MST Models Relationship Chart states that the GASM model is used to calibrate the effective gas diffusion coefficients used in the SOLM model.

The NRC staff notes that SOLM generates radionuclide inventories using a simplified decay and transmutation model with a reduced set of tracked isotopes relative to that used in standard inventory codes such as ORIGEN/SCALE. During the audit review, the NRC staff also noted that for some release fractions for Eu-154, Eu-155 and Sr-90 were being underpredicted when compared to AGR-2 data. This suggests that Eu diffusivity through intact TRISO at 1600°C is underpredicted, while Sr-90 results had cases of over and under predictions using different compact data. Given this observation, the NRC staff understands the approach being taken to model diffusivities and will verify the use of diffusion coefficients that do not underpredict potential releases of radionuclides. The NRC staff determined that the diffusion models applied appear reasonable given the preliminary approach in the information provided in the Xe-100 MST TR and audit discussions. The NRC staff will review the diffusion coefficients, including treatment of uncertainties, at the OL application stage to ensure the radionuclide inventories are well predicted.

3.1.4.1.1.4 Gaseous Fission Product Transport Model

The NRC staff reviewed the XSTERM GASM which evaluates steady-state release to birth ratio of gaseous fission products from TRISO-coated fuel particles and pebbles, including gaseous fission product releases from heavy metal contamination, particle coating failures during normal operation, and particle coating failure from heat up during LBEs.

LME noted that GASM is used to evaluate effective diffusion coefficients that are used in the SOLM module which is used to evaluate gaseous fission product release in addition to solid fission product release. As noted above for SOLM, the NRC staff will review these models in the review of a future OL application to ensure that the radionuclide inventories are well predicted.

3.1.4.1.1.5 Dust Production Model

The NRC staff reviewed the XSTERM DUSTM which describes how the dust production rate model calculates graphite and metallic dust production rates for the Xe-100 during normal operation. Graphite and metallic dust are produced during operation and radionuclides produced in the core can be transported along with dust in the coolant. To track dust-associated radionuclides, the quantity and distribution of dust must be evaluated. The dust production rate model is a key element of the evaluation of this distribution. The Xe-100 MST TR discusses the DUSTM model in section 5.1.5 and appendix E.

The NRC staff reviewed the models for dust production in the core, the fuel handling system, and the reactivity control system. **[[**

]]. The quantity of dust is calibrated against observed dust generation in previous reactors and scaled with common relations.

The NRC staff determined that the dust generation models described in the Xe-100 MST TR appear reasonable because they are based on models used for dust generation in other HTGR designs and that these models were calibrated against available data. During a future OL application review, the NRC staff will confirm that sufficient monitoring is in place to quantify the amount of dust expected to be generated during operations.

3.1.4.1.1.6 Helium Pressure Boundary Model

The NRC staff reviewed the XSTERM HPBM which evaluates radionuclide transport within the HPB during normal operation by helium flow either as free species or associated with dust particles, as well as pebble movement through the core and the fuel handling system (FHS). The HPBM considers deposition/lift off to evaluate the distribution of radionuclides after they leave the pebbles. HPBM includes physical models for sorption of radionuclides on dust particles, deposition of radionuclides on structural components and pebbles, movement of deposited radionuclides and dust particles on the fuel pebbles in the core and FHS, and re-entrainment of radionuclides and dust. The Xe-100 MST TR, Revision 3 discusses the HPBM in section 5.1.6 and in appendix F. During the audit, the NRC staff reviewed the documentation to support the generation of the HPB inventory and as supported by the NRC staff's confirmatory calculations, the NRC staff determined that fully releasing the entire HPB inventory had a negligible contribution to overall dose. The NRC staff determined that the models discussed in HPBM appropriately consider and track the radionuclides commensurate with its possible contribution to overall dose.

3.1.4.1.1.7 Core Corrosion Calculations Model

The NRC staff reviewed the XSTERM CORRM which simulates the mass transport and chemical reaction aspects of the core corrosion phenomena encountered during water or air ingress in the core of the Xe-100. During some LBEs, significant quantities of air could be introduced into the reactor vessel and interact with the fuel pebbles. Water and oxygen can react with fuel pebbles and structural graphite to potentially compromise the structural integrity

of the fuel pebbles and result in fission product release via oxidation or hydrolysis of the fuel. To mitigate this potential corrosion impact, LME designed the pebbles so that the oxidation front does not fully pass the pebble outer fuel-free zone surrounding the fueled region during postulated LBEs.

The Xe-100 MST TR describes models for oxidation of graphite by oxygen and by steam and lower-temp gas release due to hydrolysis of fuel kernels in the event that they are reached by steam. The Xe-100 MST TR, Revision 3, appendix G describes the different behavior of oxidation at high and low temperatures due to the fact that, at high temperatures, chemical reactions are faster than reactant diffusion leading to surface reactions, but, at lower temperatures, the oxidation is slower than diffusion leading to uniform oxidation within the material. The oxidation models described by LME, although the NRC staff finds them to be adequate for the preliminary analysis in the CP, do not seem to explicitly consider the penetration of oxidizer into graphite. **[[**

]. The NRC staff will review this aspect further at the OL application stage to ensure that the analysis accurately captures the penetration of oxidizer into the graphite, or, if a conservative approach is used, that sufficient justification is provided to demonstrate that it is conservative for the full range of events.

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The Xe-100 MST TR, Revision 3 discusses CORRM in section 5.1.7 and appendix G. PSAR section 6.4.2.1 references the utility of the fuel-free zone to protect particles from oxidation. If the oxidizer does not pass the fuel free zone, it does not react with SiC or exposed fuel kernels. As part of the NRC staff's review for the OL application stage, the NRC staff will review and assess the adequacy of the calculations related to the depth of steam penetration relative to the depth of the fuel free zone. The NRC staff did not focus on the fuel kernel hydrolysis or air interactions with fuel kernels and gas release for the CP review because of the reliance on the oxidizer not substantially traversing the fuel free zone.

One area in which NRC confirmatory calculations achieved different results than calculated by LME was in the amount of combustible gas generation during SGTRs. After an additional scoping calculation to identify the cause of the discrepancy, the NRC staff finds that the difference occurred primarily due to the amount of recirculation predicted by the different codes due to differences in assumptions about the available flow area and hydraulic resistance. Furthermore, the amount of water that is predicted to enter the primary system through the ruptured tube depends on the timing of isolation of feedwater and main steam lines. The possibility of recirculation and amount of water available can affect the combustible gas generation and the NRC staff will review the modeling adequacy of this event in the review of the OL application.

The NRC staff determined that the described PSAR models used to evaluate the core corrosion, which includes the effects oxidation and hydrolysis, were documented and finds them to be reasonable based on the preliminary design and associated analysis for the CP application. The NRC staff will review these models in a future OL application to ensure that the corrosion models described are justified to reflect their use including the evaluation of not significantly oxidizing past the fuel-free zone.

3.1.4.1.1.8 Tritium Production and Transport Model

The XSTERM Tritium Production and Transport Model (TRITM) is used to calculate the overall tritium mass balance in the Xe-100 design. The Xe-100 MST TR discusses the XTRIT/TRITM module in TR section 5.1.8 and indicates that the module is still under development. During the audit, the NRC staff reviewed documentation to verify how XSTERM calculates tritium releases. The NRC staff finds that the model's mass balance calculations account for the production and transport of tritium in the core, as well as the permeation of tritium into the secondary system. Tritium production within the Xe-100 design is from ternary fission, neutron activation of Helium- 3 in the primary helium coolant, neutron activation of lithium impurities in the fuel-element matrix and core graphite, and neutron capture reactions in boron used in control materials. The tritium concentration is controlled using the helium purification system (HPS) to remove tritium from the primary helium coolant. Tritium that is not removed from the primary helium coolant can escape into the secondary system by permeation through the SG. At elevated temperatures, tritium chemisorbs on irradiated nuclear graphite, therefore a large fraction of tritium is expected to be absorbed on the graphite in the core region. This results in tritium in the graphite being quickly released back into coolant if water is introduced into the core region.

During the audit, the NRC staff observed that the tritium source term is not a significant contributor to the overall accident dose consequence and agrees that the radiological consequences of Xe-100 accident releases would not be driven by tritium releases. The NRC staff also observed in the audit that LME does use the TRITM calculation for normal operational source terms. However, as a piece of the MST methodology, the NRC staff evaluated the information made available to calculate tritium concentrations within XSTERM which uses XTRIT/TRITM to perform time-dependent tritium mass balance calculations in conjunction with the XHPB module. During audit discussions, the NRC staff confirmed that the LME analysis documentation determined peak tritium after a few years of operation, and the NRC staff's review confirmed permeation through the SG tubes and HPS operations are the main mechanisms that remove tritium from the helium coolant. During the audit, the NRC staff confirmed that LME references the permeation models presented in **[**

]. The NRC staff will further evaluate XSTERM's TRITM in an XSTERM TR XSTERM TR to support the OL application.

3.1.4.1.1.9 The NRC Staff's Evaluation of XSTERM Models

The NRC staff's review of each of the XSTERM models as documented above determined that the methods contained in each model collectively represent the transport of radionuclides from the core to the environment to provide an appropriate approach for determining the LBE source terms for the Xe-100. The NRC staff determined through review of information provided in PSAR sections 3.3 through 3.6, and confirmed in audit of supporting information, that the source terms for DBA and non-DBA LBEs were developed following the approach documented in the Xe-100 MST TR and is consistent with RG 1.233 because the source terms are based upon mechanistic analysis. The NRC staff's review is based on the referenced methods, models, and assumptions captured in the Xe-100 MST TR, and the method for determining source terms for use in consequence analysis as discussed in SE section 3.1.4.1.1. In addition, PSAR section 3.1.4 states that a separate TR for XSTERM will be submitted for the NRC staff's review to support the OL application. The NRC staff will review this document as mentioned in each of the module evaluations discussed above.

3.1.4.1.2 Radionuclide Inventories used in LBE analysis

PSAR table 3.3-2 provides the initial radionuclide isotopic inventory for all LBEs unless otherwise noted in an LBE description. The NRC staff examined the calculation of the radionuclide inventory provided in PSAR table 3.3-2 during the audit and noted that some aspects of the calculation appear to be non-conservative (e.g., the assumed fuel failure fraction is similar to the best estimate value obtained for irradiated AGR-1 and AGR-2 testing as reported in EPRI-AR-1(NP)-A (ML20336A052), but less than the 95th confidence value).

The NRC staff compared the radionuclide inventory provided in PSAR table 3.2-2 to the radionuclide inventory from the PBMR-400 benchmark calculation and the radionuclide inventory from Xe-100-specific independent NRC calculations and finds that the inventories in table 3.2-2 are approximately five orders of magnitude smaller than expected due to the fact that the table reports the radionuclide inventory in the HPB, whereas the dominant source of possible dose in an accident occur from the release of radionuclides in the pebbles, which is consistent with the NRC independent calculation.

Although the NRC-evaluated pebble inventories were closer to the pebble inventories that LME provided in support calculations, the values did not completely match. The NRC staff audited the review of the approach that LME uses to evaluate radionuclide inventories within XSTERM, including isotopic yields, decay chains, and neutron capture transitions. Although LME-calculated SCALE inventories generally matched NRC-calculated SCALE inventories, LME's XSTERM-calculated inventories (used for dose) returned substantially different inventories for the same isotopes. XSTERM uses a reduced set of isotopes in its evaluation of inventories since it focuses on dose significant isotopes and isotopes that can decay or transmute into dose significant isotopes.

During the audit, the NRC staff reviewed a list of radionuclides that were part of the radionuclide inventory calculations for the MST analyses. The NRC staff confirmed that the list provided by LME represents those radionuclides contributing 1 percent or more to the total dose. During the audit, the NRC staff reviewed documentation that compared XSTERM radionuclides to the larger list of radionuclides provided by SCALE and noted that additional radionuclides were described as being significant enough for inclusion outside of the current set described by XSTERM for the CP. This documentation provided details to confirm the selection process of radionuclides and identified additional radionuclides that will be addressed in the OL application. The NRC staff anticipates that these additional radionuclides will not significantly impact dose but will confirm this during the review of a future OL application.

As stated in PSAR section 3.1.4, LME will provide more detailed information on XSTERM and its implementation in calculations supporting the PSAR in a TR that is planned to be submitted for the NRC staff's review that would support a future OL application. In addition, the NRC staff reviewed the HPB radionuclide inventories contained in PSAR table 3.3-2 and confirmed during the audit that the HPB inventories were determined consistent with the Xe-100 MST TR. During the review of a future OL application, the NRC staff will review the XSTERM documentation which includes any updates to the XSTERM modeling of radionuclide inventory.

3.1.4.1.3 Source Terms used in LBE analysis

The NRC staff reviewed the applicant's processes for determining the scenario specific released source terms in radionuclide release tables provided in PSAR sections 3.3 through 3.6. The tables provide activities released to the environment: from fuel during heat up, from the airborne HPB "steady state inventory," liftoff of radionuclides deposited on HPB surfaces including radionuclides associated with dust, steam off of radionuclides deposited on HPB surfaces, wash off of radionuclides deposited on HPB surfaces, and corrected 4-day, 30-day released activities from pebbles, and total release activities from pebbles and from the combination pebbles and HPB. The correction factor in these tables accounts for differences between two XSTERM code versions. The audit allowed the NRC staff to review the calculations supporting the activity releases reported within the PSAR tables and to confirm the consistent use of the proposed MST approach. The NRC staff notes the following about the listed activity release contributions:

- The activities provided under fuel heat up reflect the additional release out of the fuel during the initial plant transient portion of the event.

- The activities under "steady state inventory" reflect the expected releases during an accident from the HPB gas space during normal operations as noted in PSAR section 3.1.5.

- The activities under liftoff reflect the deposited or plated out graphite dust on HPB surfaces becoming detached due to increased flow rates due to a break or opening in the HPB as discussed in PSAR section 3.1.5.

- As described in PSAR sections 3.1.4 and 3.6, the activities under steam off and wash off reflect contributions from radionuclides plated out on reactor surfaces becoming detached when steam or liquid water is introduced to the HPB. These activities also reflect the calculated activity contributions from failed particle hydrolysis and oxidation releases due to steam or air ingress events.

- The activities provided under 4-day, or 30-day corrected activity reflect the long-term event-specific activity determined for releases from fuel and may account for reduction factors for plate-out and hold-up. These activities reflect the activities released after the initial plant transient for the time periods specified.

- In total, the activities from fuel heat up, steady-state inventory, liftoff, steam off, wash off, and 30-day corrected activities contribute to the 30-day source term available for release for each event.

- Total pebble released activity reflect the total activity from both fuel heat up and 30-day corrected activity.

For scenarios that do not involve an HPB break or lift of safety relief valves, the short-term releases (steady-state inventory, lift off, steam off, and wash off) are set to 0.

The NRC staff determined that the radiological releases calculated using XSTERM are consistent with the methodology described by the Xe-100 MST TR and are representative of the preliminary Xe-100 design described in the PSAR as noted above in this subsection of the SE. Planned development activities for XSTERM were discussed during the review, along with V&V plans to ensure a qualified code representative of the LMGS design will be available to support the OL application. PSAR section 3.1.4 states that LME will provide more detailed information on XSTERM in a licensing TR for the OL application.

3.1.4.1.4 Non-core related source terms

The NRC staff reviewed PSAR section 3.10 which describes the non-core related sources for LMGS. The NRC staff's evaluation of the events is provided in SE section 3.10.

LME provided information on the fuel sources (fuel handling accidents) and non-fuel sources located outside of the HPB to support the NRC staff's confirmation of the information in PSAR section 3.10. The NRC staff reviewed the associated source term calculations which facilitated the NRC staff's understanding on the method used for determining the LMGS fuel pebble inventory for fuel handling events. The applicant used VSOP to generate inputs for SCALE which was then run to provide a fuel pebble inventory. The results of this fuel pebble inventory were confirmed by NRC independent calculations discussed in SE section 3.1.4.1.5. The NRC staff's confirmatory calculation using SCALE demonstrated that the fuel pebble inventory determined for use in fuel handling accidents appears appropriate and consistent with inputs used for determining source terms using XSTERM.

PSAR section 3.10 describes the assumptions used to generate source terms for non-core and non-fuel related events. Source terms regarding liquid and solid waste are dependent on the amount stored at any given time. The NRC staff compared these assumptions to the preliminary estimates provided in PSAR chapter 9, "Control of Routine Plant Radioactive Effluents, Liquid Radwaste, Solid Radwaste, and Plant Contamination." The NRC staff finds the solid waste source term in PSAR section 3.10, bounds the estimates provided in PSAR table 9.3-2, and the results for liquid waste represent a year's worth of liquid radwaste from one module. The NRC staff notes that LMGS plans monthly shipments of liquid radwaste stored onsite to an offsite location; therefore, this quantity of liquid radwaste is not expected onsite during plant operations involving four modules. The NRC staff finds this source term appropriate since it bounds the expected radwaste stored onsite. When accounting for further reductions in source terms as described in PSAR 3.10.4.2, the source term is then further reduced when accounting for the fact the waste is in solid or liquid form, with the potential of the waste being prepackaged and subsequent release as an airborne material would result in significant reductions to the radionuclides released when assuming small fractions of liquid would be released as airborne material since an accidental release of liquid would not likely convert to an airborne release. Additionally, the NRC staff's review of the other described non-fuel sources finds that the applicant provided justifications regarding assumed inventory in discussions or PSAR tables in PSAR section 3.10.

The NRC staff determined that the radionuclide inventory for non-core related sources is calculated without the use of XSTERM, but with inputs that are consistent with those used in XSTERM source term calculations. The NRC staff's review determined that the source terms described in PSAR section 3.10 are based upon conservative estimates of the quantities of material expected to be present for the identified non-core or non-fuel events. Based upon the justifications provided in the PSAR, the NRC staff finds that the source terms described reasonable for the evaluation of radiological consequences.

3.1.4.1.5 NRC Staff Confirmatory Analyses

Although a review of physical models can be used to confirm the consistency with known physical models, an independent assessment that returns similar results can provide confidence that the combination of models, along with the initial conditions and boundary conditions chosen for the problem, are providing reasonable results. Independent assessments can be used to identify areas which need further review, when they do not align. Given the complexity of the involved modeling, independent assessments such as these allow the ability to assess the validity of models and applications. In order to confirm the radiological releases for different accident scenarios, the NRC staff conducted analysis of the radionuclide inventory using SCALE and accident simulations using MELCOR to provide independent analyses of transient temperature behavior and radiological release estimates.

The majority of the physical models for SCALE and MELCOR were developed and implemented for prior work, therefore, the models were adjusted to apply to the Xe-100 design as described in the PSAR. Some of the earlier developed SCALE and MELCOR HTGR physical models and implementations are described in the following documents:

- "PBMR Coupled Neutronics/Thermal-hydraulics Transient Benchmark - The PBMR-400 Core Design," NEA/NSC/DOC(2013)10, July 2013

- Bostelmann, F., et al., "SCALE Capabilities for High Temperature Gas-Cooled Reactor Analysis," Annals of Nuclear Energy, Vol. 147, 107673, November 2020.

- M. Young, et al., MELCOR Fission Product Release Model for HTGRs, SAND2010-0800C

- K. Wagner, B. Beeny, and D. Luxat, MELCOR Accident Progression and Source Term Demonstration Calculations for HTGR, SAND2022-2750, Both R1: 2022 and R2:2024.

The NRC staff's MELCOR analyses focused on accident scenarios that are generally bounding release scenarios. The full scope of confirmatory analyses was selected to cover a range of phenomena and support confirmation of each aspect of the source term calculations. The MELCOR analyses also independently assessed the impact of parameter uncertainty by comparing upper bounds of input values to nominal conditions. The primary scenarios that the MELCOR analyses focused on were DLOFC scenarios and SGTR scenarios. DLOFC scenarios were modeled because they have the highest temperatures, failure of the HPB barrier, and a possibility for air ingress which can result in fuel oxidation, causing them to be some of the highest release scenarios. SGTR scenarios have the potential for steam oxidation of fuel which can drive larger releases and exercises unique phenomena within the codes. The MELCOR analyses also included other scenarios to check specific PSAR results (differences in break size for HPB depressurization, control rod group withdrawal (CRW), and pressurized loss of forced cooling (PLOFC)).

These analyses provide independent verification of radionuclide inventories, end-of-life and accident scenario temperature distribution histories, fuel failures, and fission product releases that contribute to dose. In some cases, the MELCOR analyses used the same physical models used in the XSTERM analyses (core-wide heat transfer, conductivities, and fission product diffusivities). In cases in which the same physical models are adopted, rather than different or independent models, the confirmatory calculations are primarily to check the implementation of the models rather than to serve as a fully independent check on the calculation.

The MELCOR analyses performed by the NRC staff focused on temperature distributions at the end of 60 EFPY and for accident scenarios: temperature distributions in TRISO, pebbles, pebble beds, reflectors, and RCCS; failures of TRISO particles, transport and distribution of fission products to environment with evaluation of dose. Steady-state temperature distributions and temperature distribution histories during accidents are affected by the thermal conductivities for pebble and reflector graphite grades that change with both temperature and neutron fluence. The NRC staff note that differences in fluence can result in large drops in conductivity which lead to increases in temperature since higher temperatures result in higher predicted fuel failure rates, degradation of fission product barriers, and fission product diffusion rates through intact barriers, all of which contribute to radiological releases to the environment and thus dose.

NRC staff compared radionuclide inventories, decay power curves, power distributions, and reactivity coefficients calculated by NRC staff using SCALE to those provided by the applicant in the PSAR and supporting calculations. The NRC staff compared the temperature distributions during operation and temperature distribution histories during accident scenarios calculated with MELCOR analyses to results obtained from Flownex, GOTHIC, and XSTERM.

Radionuclide inventories differed between the PSAR and the NRC staff SCALE results. Through additional discussion with the applicant in audit, NRC staff were able to determine that

differences in radionuclide inventory were primarily due to missing production and loss terms in XSTERM caused by the limited number of isotopes tracked. Most of the impacts were small with the exception of I-132 which was significantly underpredicted in the PSAR source terms. This underprediction has since been addressed in an updated version of XSTERM and the NRC staff performed an assessment of the overall impact of this error on the dose consequence results. While it did impact the fraction of the dose coming from iodine it did not have a significant impact on the overall dose due to the other isotopes still being the dominant dose contributors and there being additional margin within the preliminary calculations. NRC staff will review results of the V&V for these updated portions of XSTERM at the OL application stage.

Although reported radionuclide inventories differed, the power distributions, decay power curves, and reactivity coefficients evaluated by the NRC staff using SCALE generally aligned with those provided by the applicant in the PSAR. The results were used by the NRC staff to analyze the impact of the influence of dose on reflector conductivity, RCCS performance, exploration of different thermal conductivity, and fission-product diffusivities in different materials. The NRC staff determined the MELCOR results for the most part returned similar operating temperature distributions and accident temperature histories for equivalent initial conditions and boundary conditions.

The NRC staff's independent calculation of these parameters identified areas where additional clarification in the PSAR was needed and areas where, since similar results were being obtained independently, greater confidence in the PSAR results can be found. Due to the complexity of models contributing to dose, obtaining confidence in an applicant's results is difficult without conducting confirmatory independent analyses since it is not possible to ensure that the physical models presented in the report match those implemented in the code and that the implementation of the models and boundary conditions provide reasonable results.

Examples where the independent NRC calculations and review of materials in audit led to the improvement of understanding of the modeling of phenomena include: radionuclide inventories, initial temperature distribution resulting from required assumptions of neutron fluence in the reflector, the influence of diffusivities on releases, and the influence of recirculation on hydrogen generation during SGTR scenarios.

The MELCOR analyses were also used to explore the impact of variabilities in input parameters. The MELCOR analyses analyzed the impact of variation in not only the PKPIRT parameters from the PSAR for different cases (e.g. power, decay power, thermal conductivities, emissivity), but also the influence of expected variations in initial conditions based on potential normal operating ranges for key parameters (helium flow, inlet temperature, main steam pressure, feedwater mass flow, feedwater temperature, and reactivity coefficients).

Some results of the MELCOR confirmatory calculations are shown in section 3.6 of this SE including some discussion about the variations duplicating the PKPIRT and additional parameters.

Iterative confirmatory calculations were conducted to verify LMEs temperature calculations. Initial MELCOR confirmatory modeling showed lower temperatures for accident scenarios, including the DLOFC. The differences were traced by the NRC staff to the effects of neutron damage on graphite and alignment of RPS signal timing, SG heat removal, and break size and flow. The MELCOR temperatures aligned with those of the PSAR following changes made to the analysis by the NRC staff upon receipt of the additional information on the range of fast

fluence irradiation in graphite and on the other parameters. Figures 3.1-1 and 3.1-2 show the initial and updated DLOFC temperature profiles for the confirmatory calculations, respectively. SE figure 3.1-2 for the updated calculation also includes PKPIRT variations.

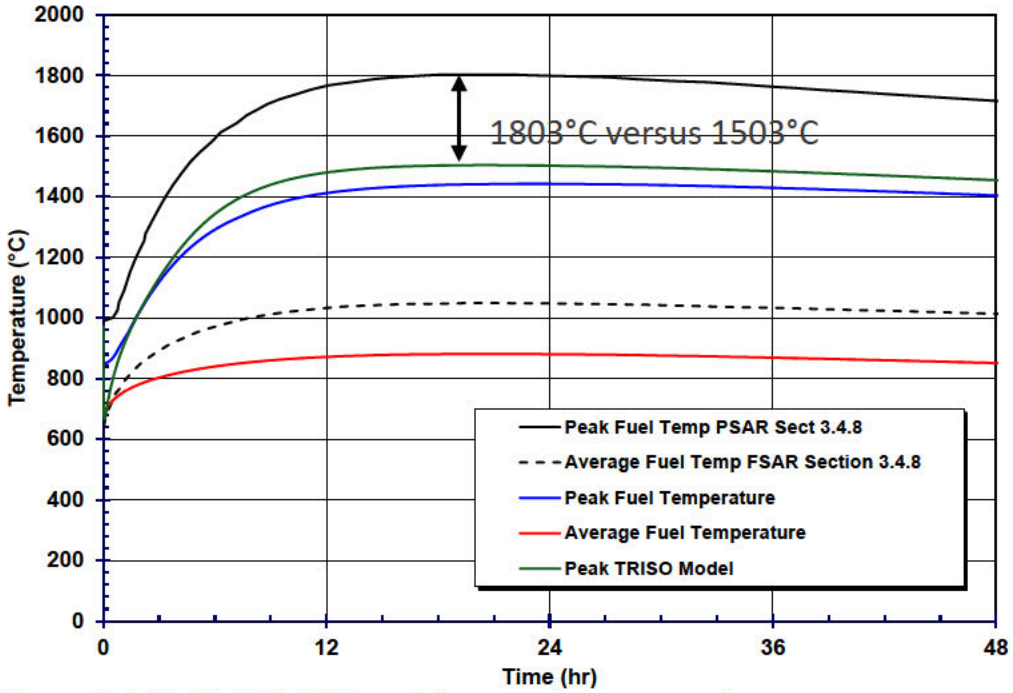

Figure 3.1-1 Initial DLOFC peak temperature comparison

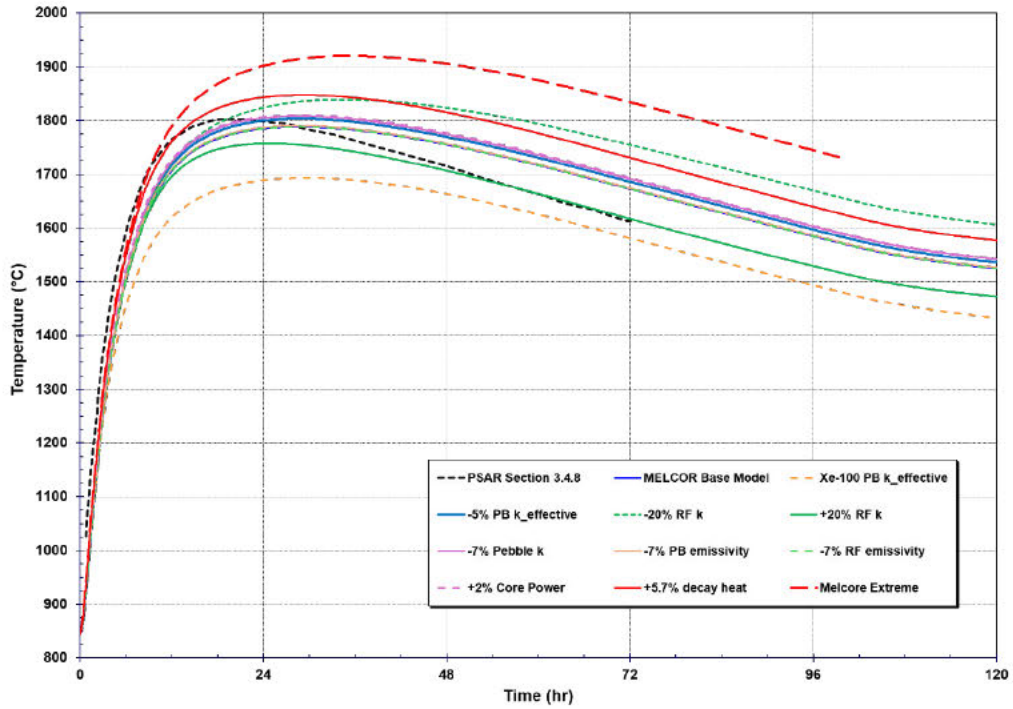

Figure 3.1-2 Updated DLOFC peak temperature comparison along with reproduction of PKPIRT parameter variations

NRC confirmatory calculations also evaluated the impact of initial conditions related to potential operating parameter envelopes, including helium mass flow, reactor inlet temperature, feedwater flow rate, SG pressure, feedwater temperature, and increased range in variability of reactivity coefficients on the results. Based on the results of these sensitivities the NRC staff confirmed that these additional variations did not significantly impact the temperature profile.

The NRC staff's initial confirmatory calculations for the SGTR DBA returned substantially more hydrogen generation than LME's calculations. The MELCOR confirmatory calculations were used to track the source of the difference to recirculation flow in the HPB loop. SE figure 3.1-3 shows the flows with and without reverse flow. SE figure 3.1-4 shows the resulting hydrogen generation for the same conditions. The confirmatory calculations predict hydrogen generation similar to that in the LME calculations if recirculation is prevented. The main difference in assumptions between the LME and NRC/SNL analyses that led to the difference was the flow resistance of the circulators: LME analyses assumed a high resistance whereas the NRC analyses assumed no resistance (i.e., circulators not modeled). The NRC staff note that this scenario involves an assumption of failure of HPB circulator check valves that would prevent this reverse recirculation flow. The assumptions are described in PSAR section 3.6.9.1.

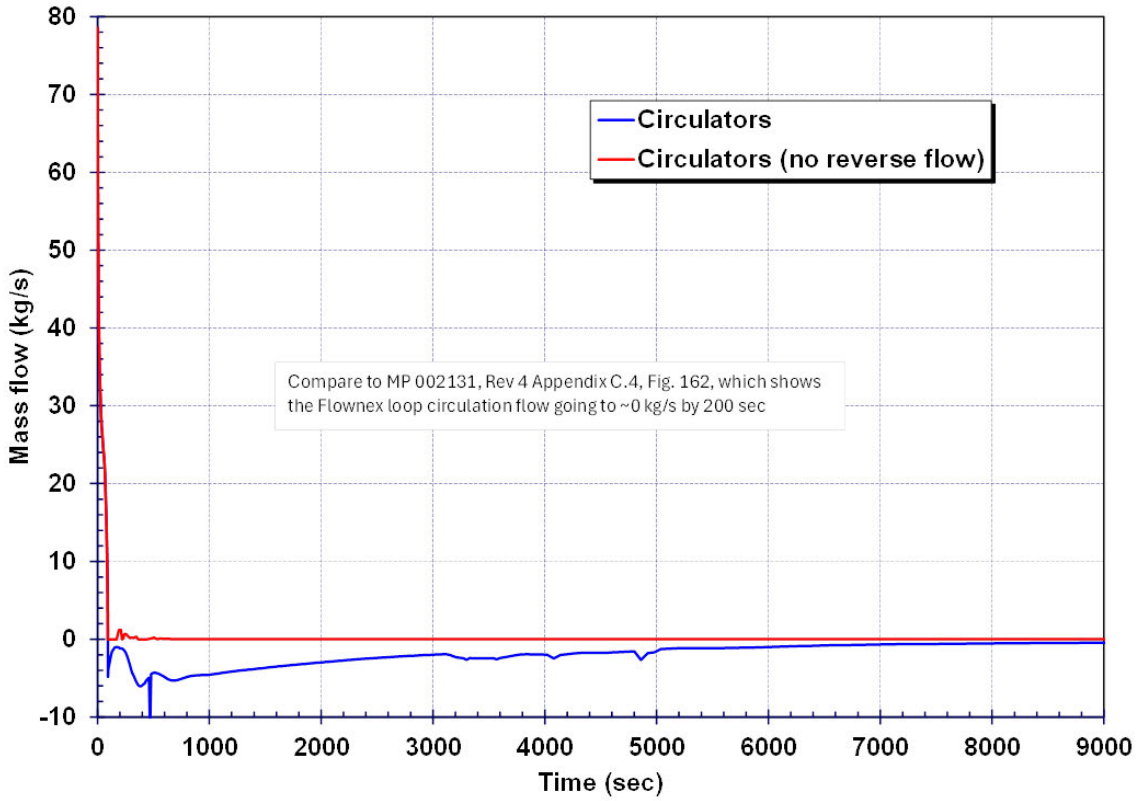

Figure 3.1-3 Confirmatory MELCOR flow with reverse flow allowed and with reverse flow blocked

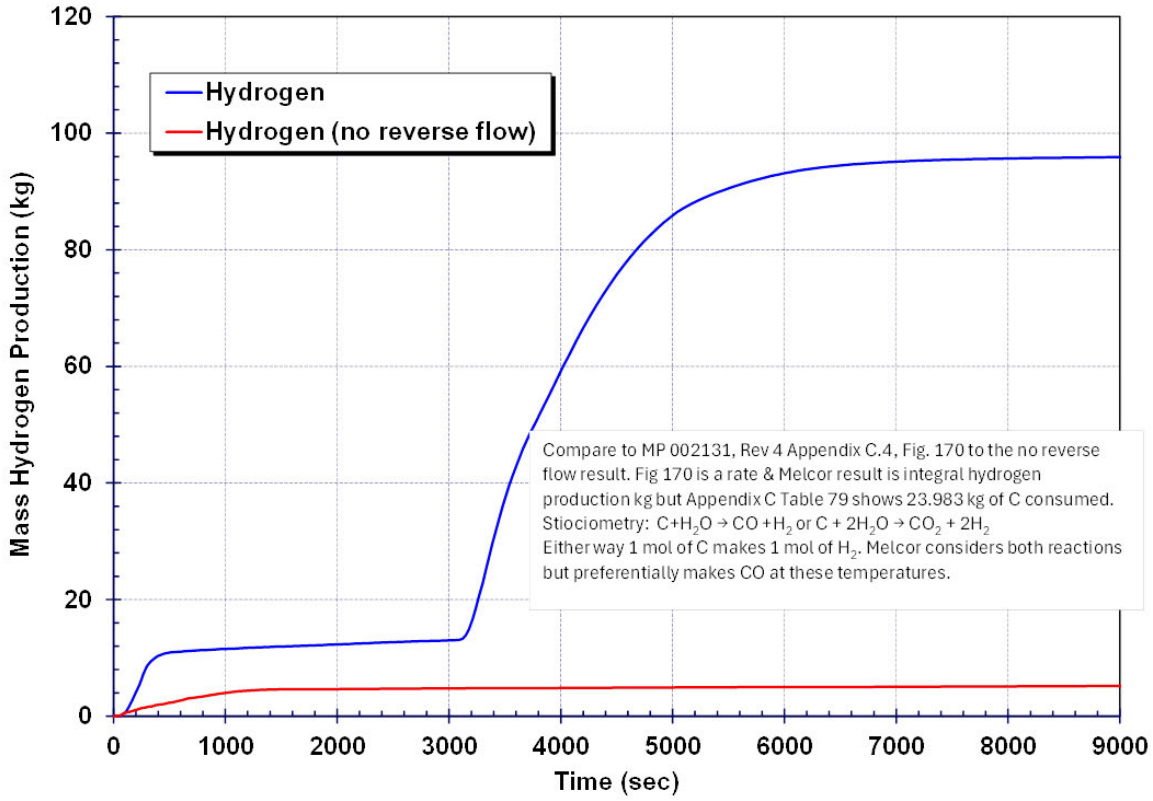

Figure 3.1-4 Confirmatory calculations showing the influence of reverse recirculation on hydrogen generation

Major findings from the confirmatory calculations include:

- Graphite properties are important to overall accident behavior and have a significant impact on fuel temperature profiles as they change with neutron fluence, temperature, and porosity (manufacturing).
- Results confirm the need to replace some of the reflector graphite due to property degradation from neutron fluence.
- Assessing the graphite damage is complex and requires an iterative approach.
- Available radionuclide diffusivity information is limited. With limited exceptions, LME values seem to be conservative relative to other sources for these diffusivities.

The NRC staff calculations generally align with calculations provided by LME on power distribution, reactivity feedback coefficients, and temperature distributions for different LBEs given similar input assumptions. Based on the information summarized above the NRC staff's confirmatory calculations provide confidence in the preliminary results reported for the CP application. These key topics will be evaluated further during the review of a future OL application when additional graphite testing and code V&V information is available.

3.1.4.1.6 Uncertainty Applied to the Source Terms used in LBE analysis

To address uncertainty in the source term analysis, the applicant describes the use of a PKPIRT and V&V of the XSTERM computer as described by the Xe-100 MST TR. LME describes the use of the PKPIRT process to identify key parameters that should be varied to achieve 95th percentile values in their source term calculations. The key parameters vary by event but generally include thermal properties (e.g. for DLOFC: pebble effective reactivity, fuel sphere conductivity, graphite thermal conductivity, pebble bed emissivity, reflector emissivity, power, and decay heat) and reactivity feedback coefficients. Distributions on source-term specific information, e.g. fission-product diffusivities in different materials, were not included in the uncertainty analysis. The NRC staff finds that the description of the PKPIRT process did not provide a sufficient basis for demonstrating the results are representative of 95th percentile values. In PSAR section 3.1.4 the applicant states that for the OL application, comprehensive uncertainty quantification will be performed considering all the facets of MST uncertainty, including code bias. The NRC staff finds this acceptable because full treatment of uncertainties is typically provided at the OL application stage. During the review of a future OL application, the NRC staff will evaluate the development of the FSAR source terms to ensure that the effects of diffusivities, fuel failure fractions, and matrix and reflector conductivities are addressed and that the uncertainty analysis demonstrates an appropriate level of conservatism is achieved. The NRC staff also highlights that the Xe-100 MST TR describes the validation process for the XSTERM as consistent with the approach described in RG 1.203. RG 1.203 describes an acceptable approach for assessing EMs which includes describing the V&V activities that will be performed for uncertainty analysis.

3.1.4.1.7 Evaluation of Barriers and Functional Containment

As described in SECY-93-092, a MST is “the result of an analysis of fission product release based on the amount of cladding damage, fuel damage, and core damage resulting from the specific accident sequences being evaluated. It is developed using best estimate phenomenological models of the transport of the fission products from the fuel through the reactor coolant system, through all holdup volumes and barriers, taking into account mitigation features, and finally, into the environs.”

SECY-18-0096, which was approved by the Commission, describes the establishment of functional containment performance criteria for non-LWRs and provides a high-level definition of functional containment as “a barrier, or a set of barriers taken together, that effectively limits the physical transport of radioactive material to the environment.” Enclosure 2 to SECY-18-0096 describes a proposed technology-inclusive, risk-informed, performance-based approach to derive functional containment performance criteria.

The Xe-100 MST TR describes that the Xe-100 reactor technology employs a functional containment consisting of a series of barriers to radionuclide transport and lists these barriers as the TRISO fuel, the reactor core graphite and other carbonaceous materials, the HPB, and the RB.

The NRC staff reviewed PSAR section 3.1.4 which describes five successive barriers to fission product radionuclide release that contribute to the functional containment of radionuclides: (1) UCO fuel kernel, (2) TRISO coatings, (3) pebble fuel matrix, (4) HPB, and (5) the shield structure (SST) in the RB which is not being credited for radionuclide retention. PSAR figure 6.4.2-3 provides additional information on the Xe-100 functional containment approach. NRC

staff finds this description of barriers consistent with the Xe-100 MST TR because they identify the same barriers and clarify that radionuclide holdup in the RB SST are not credited. Additionally, the NRC staff notes the functional containment approach is captured by PDC-RFDC and CDC 16, which the NRC evaluates in SE section 5.3.

The NRC staff review of the release mechanisms of the UCO fuel kernel leveraged information from AGR-2 testing data found in INL/RPT-23-74651, "AGR TRISO Fuel Fission Product Release Data Summary," and information made available for NRC staff review during the audit. During the audit, the NRC staff reviewed several XSTERM reports that support the validation efforts of the XSTERM computer code and provides sources of fractional release information from AGR-2 compact experimental data which leveraged both UCO and UO_2 fuels. LME provided their XSTERM validation report which showed the comparison of AGR-2 compact release fractions to the results of XSTERM. Based on the information provided during the audit, the NRC staff observed that XSTERM was overestimating the release to birth ratio during irradiation tests, and during annealing the amount of fractional release was generally higher when compared to the AGR-2 experimental data.

The NRC staff review of TRISO coatings confirmed the XSTERM described release mechanisms for the IPyC coating, the SiC layer, and the Outer Pyrolytic Carbon (OPyC) coating. The assumptions for these release mechanisms include the manufactured fuel failure fractions and fuel failure fractions (FFF). The NRC staff notes that PSAR section 3.3 specifies the allowable initial FFF of 1.1×10^{-4}, which is the maximum value acceptable for the Xe-100 fuel specification. As discussed in SE section 3.1.4.1.1.2 XSTERM's FPM calculates particle failure which models and calculates SiC pressure vessel failure due to internal gas pressure in the particles, kernel migration, fission product corrosion, thermal decomposition, and IPyC cracking.

The NRC staff review of the pebble fuel matrix confirmed its inclusion of diffusion as part of the total diffusion modeled under SOLM as discussed in SE section 3.1.4.1.1.3, which would account for the diffusion through all layers of the TRISO pebble (IPyC, SiC, OPyC and the graphite sphere matrix). In addition, the pebble fuel matrix has a fuel free shell which acts to limit radiological releases during a SGTR which would include oxidation and hydrolysis releases on the fuel pebbles but also release deposited material on the HPB. The presence of this fuel free layer was also a key difference between the LMGS fuel and the AGR compacts, resulting in additional radionuclide retention and lower release fractions than seen in the AGR test results. To confirm that the observed reduction was reasonable, the NRC staff reviewed sensitivity runs and intermediate XSTERM results in audit. Validation of XSTERM between CP and OL application stages will confirm that the models used to account for the impact of the fuel free layer are consistent with available test data. NRC staff will review these validation results at the OL along with available results from XSTERM demonstrating the source term impacts.

The HPB serves as the secondary barrier for radionuclide retention and as stated in PSAR section 1.3.2.1, is necessary for DID adequacy but not required to meet required safety function 1, "retain radionuclides in fuel particles and pebbles." During normal operations the distribution of radionuclides, including those attached to graphite dust, is transported through the HPB. During events such as a SGTR, steam and air ingress is expected to release deposited material on HPB surfaces around the break to contribute radionuclide releases from expected steam off and wash off inventories to the event-specific source terms. Releases from the HPB would enter the RB SST. For depressurization events, once the pressure reaches equilibrium with the environment, further release is limited to "breathing" of the HPB. LME performed a series of runs to evaluate the fraction of release that would occur following equilibrium being reached and

selected what LME asserted was a bounding value of 4 percent for the preliminary analysis reflected in the PSAR, meaning the HPB is providing 96 percent retention of radionuclides released following the initial transient for some events. The NRC staff review determined that this method of calculating HPB retention using the thermodynamic codes selected is reasonable. Furthermore, the use of a single value to bound others is reasonable for the CP. With LME's plan to generate LBE specific source terms for a future OL application, the impact of the HPB for individual events should be available for review at the OL application stage.

The NRC staff review of the RB SST barrier notes that for specific events such as those resulting in failure of pressure relief valves to close or rupture disks burst into the SST, the CP consequence analyses take no credit for the holdup, decay, or deposition in the SST for the consequence analysis. The NRC staff review determined that the method of not accounting for RB retention is conservative because accounting for any amount of holdup, decay, or deposition will result in lower calculated doses. In addition, the NRC staff review of consequences related to venting to the RB SST notes that while this barrier is not being credited in any CP consequence analysis, it could be leveraged in future updates provided in a future OL application if LME determines that credit for this barrier is needed and supported by the final design. However, with blowout panels designed to protect the SST from damage due to rapid pressure relief, a change would likely be needed to balance the pressure relief function with any radionuclide retention function and ensure both could be achieved for the credited LBEs.

In addition, the NRC staff notes that PSAR sections 6.4.2 and 13.3.3 state that fuel testing is needed to confirm burnup, fast fluence, and temperature outside the validated ranges from previous testing will not challenge the fuel's ability to perform its RSFs. The NRC staff's evaluation of fuel testing and design aspects is provided in section 6.4.2 of this SE. The NRC staff finds that this additional R&D testing will verify the anticipated radionuclide release rates from fuel at the upper end of the temperature ranges expected for their DBAs. Consistent with 50.34(a)(8) the NRC staff finds that this is reasonable to resolve this safety question prior to the date of the completion of construction and supports issuance of the CP in accordance with 50.35(a)(3).

Based on the review of barriers and functional containment described above, the NRC staff finds that the LMEs approach to functional containment is consistent with the approach described in SECY-18-0096 by providing a high-level definition of the functional containment barriers applicable to the Xe-100 design.

3.1.4.2 Conclusion

Based on the information provided in PSAR section 3.1.4 and the MST TR IBR into the LMGS PSAR, the NRC staff finds that the MST methodology is acceptable for the preliminary analysis of the LMGS. This finding is based on the following:

- The applicant's described models for MST are appropriately used and described in supporting documentation for the preliminary analysis of the LMGS consistent with the guidance in RGs 1.233 and 1.253 and NEI 18-04 and 21-07 (see section 3.1.4.1.1 of this SE);

- The applicant's methods for determining the radionuclide inventory and source term in its MST evaluations were independently confirmed by NRC confirmatory analyses (see sections 3.1.4.1.2 through 3.1.4.1.5 of this SE);

- The applicant's treatment of uncertainty in MST evaluations is acceptable for the CP review (see section 3.1.4.1.6 of this SE); and

- The applicant's use of functional containment is reflected in the development of the MST approach and therefore acceptable (see section 3.1.4.1.7 of this SE).

The NRC staff reviewed the event-specific MST information described in PSAR section 3.1.4 and determined that the information is consistent with the guidance in RG 1.233, RG 1.247, and RG 1.253, and used an approved methodology in their MST TR (ML25073A098) to develop MSTs for LME. The NRC staff determined that the information provided in the PSAR regarding the use of MSTs in the analysis of LBEs: (1) is consistent with the guidance in RG 1.253 and NEI 21-07; (2) provides a comprehensive and technically acceptable description of the technical basis for the event-specific MSTs used in the preliminary safety analysis; (3) identifies the applicant's plans for finalizing the source terms prior to submitting the OL application; and (4) supports the safety conclusions made in SE chapters 3 and 4. Therefore, the NRC staff determined that the use of the MSTs described in the PSAR supports the issuance of a CP pursuant to the regulations of 10 CFR 50.34(a)(1), and (4), as applicable.

3.1.5 Atmospheric Dispersion and Dose Calculation Methodology

PSAR section 3.1.5 states that dispersion factor and dose results are used to demonstrate the design meets frequency-consequence targets defined in NEI 18-04. The PSAR references 007116, "Xe-100 Licensing Topical Report Atmospheric Dispersion and Dose Calculation Methodology," Revision 2 (ML23268A454). PSAR section 3.1.5 also lists inputs relevant to source term analysis which are evaluated as part of the MST evaluation provided in section 3.1.4 of this SE.

The regulatory evaluation provided in section 3.1.4 of this SE is applicable to this section.

3.1.5.1 Technical Evaluation

The NRC staff evaluation of TR 007116, "Atmospheric Dispersion and Dose Calculation Methodology," Revision 2 (ML24242A251) concludes that it is an acceptable methodology to calculate the downwind dispersion and resulting doses at the EAB, the low population zone (LPZ) outer boundary, and in the control room for the LBEs for the Xe-100 reactor subject to two conditions. L&Cs associated with TRs that are IBR, are addressed below.

The NRC staff's SE for the TR imposed the following L&Cs:

1. The applicant applying the TR methodology for DBA analysis for the EAB dose must show that the dose calculated by the methodology is bounding for any 2-hour period in order to show compliance with the offsite-dose-related safety analysis report requirements.

2. A condition of the use of generic dispersion factors calculated by the TR methodology in an analysis is that an applicant for a specific site referencing this TR will perform a dispersion analysis with site-specific data to either: 1) affirm that the Xe-100 analysis results using these generic dispersion factors are conservatively bounding, or 2) provide site-specific analysis results using dispersion factors developed during site characterization.

The applicant addressed these L&Cs in PSAR table 1.4-9.

1. The applicant provides the assumptions used to ensure that the EAB dose is bounding for any 2-hour period by ensuring the use of conservative breathing rates during the first 24 hours and follows guidance in RG 1.183 Revision 0 for 1-30 days, conservative atmospheric dispersion factors, and assumes no decay occurs for each event resulting in a release. Given that releases are dominated by early phases of the accident and no expectation for a delayed release, the NRC staff finds that the proposed assumptions are conservative for determining a bounding 2-hour dose given the use of conservative assumptions discussed above.

2. The applicant states in PSAR table 1.4-9 that they will install meteorological instrumentation and collect site-representative data in accordance with RG 1.23. In addition, the applicant states that a site-specific atmospheric dispersion analysis using this data will be performed for the OL application. This method includes confirmation that Xe-100 analysis results using the CP's generic dispersion factors are conservatively bounding or updating the LMGS licensing basis with site-specific dispersion factors.

Based on the above discussion, the NRC staff determined LME adequately addressed all the L&Cs included in the atmospheric dispersion and dose calculation methodology TR. Additional consideration for the dose consequence methodology is discussed below.

During the review of the PSAR, the NRC staff identified that the methodology used to determine the dose consequence results did not fully account for the uncertainty in atmospheric dispersion, because LME used the mean (non-DBA) dispersion factor to calculate portions of the 95th percentile dose consequences. Through audit discussions, the NRC staff confirmed that the offsite dose consequences reported in PSAR chapter 3, "Methodologies, Analyses, And Licensing Basis Events," applied a generic atmospheric dispersion factor (χ/Q) of 1.89×10^{-4} s/m^3 to all mean anticipated operational occurrences (AOOs), design-basis events (DBEs), and BDBEs. In addition, a generic DBA dispersion factor (χ/Q) of 3.57×10^{-4} s/m^3 was used to calculate the standard deviation of dose consequence profiles for similar LBEs sharing a common long-term end state, which was then used to calculate the 95th percentile dose consequence values for the long-term cooling dose constituent of the consequence results. The NRC staff also confirmed that other portions of the dose consequence constituents (e.g., helium circulating inventory, lift off, steam off) did not have uncertainty applied to them based on LME's assessment of the conservatism in the calculations and their contribution to the total dose consequence results relative to the long-term dose constituent. The uncertainty bands on the dose consequence results are used in the safety classification of SSCs and should be representative of the 95th percentile or greater dose consequences for the site, as discussed further in the following paragraph.

To address that all portions of the dose consequence results bound a 95th percentile dose consequence for the site, LME confirmed via RCI-2 (ML26258A017, ML26260C204), that the minimum χ/Q used in calculation of the 95th percentile consequences (1.89×10^{-4} s/m^3) is expected to bound the site-specific 95th percentile χ/Q. This assertion by LME is based on preliminary results for a site-representative 95th percentile χ/Q of 7.09×10^{-5} s/m^3. The NRC staff confirmed through RCI-2 that the calculation of this dispersion factor (χ/Q) was performed using the ARCON96 code in accordance with RG 1.249, "Use of ARCON Methodology for Calculation of Accident-Related Offsite Atmospheric Dispersion Factors," Revision 0 (ML22024A241), RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability

Assessments at Nuclear Power Plants," Revision 0 (ML031530505), RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1 (ML003740205), and under LME's QAP. This preliminary calculation used meteorological data from nearby weather monitoring observation sites, but LME committed to collect site-specific meteorological data in accordance with RG 1.23 to support the calculation of site-specific dispersion factors (χ/Q's) that will be provided as part of the OL application. Given this confirmation and awareness that meteorological instrumentation has been installed onsite, the NRC staff determined that the non-DBA χ/Q of 1.89×10^{-4} s/m^3 would bound the proposed site-specific 95th percentile dispersions factor and therefore bound the 95th percentile dose consequence results reported in the PSAR and F-C chart. Based on the preliminary site-representative χ/Q calculated in accordance with RG 1.249, RG 1.194, and RG 1.145, and the margin between this value and the minimum χ/Q used in calculating the 95th percentile dose consequence results, the NRC staff determined that the 95th percentile dose consequence results provided in PSAR chapter 3 are representative of the 95th percentile or greater dose consequences for the site and are appropriate for use in identifying risk-significant LBEs and in the classification of SSCs.

The NRC staff confirmed through the audit that the analyses used methodology tools, inputs, and assumptions consistent with the atmospheric dispersion and dose calculation methodology TR and used source terms as presented for various events provided in PSAR sections 3.3 through 3.6. Consistent with the atmospheric dispersion and dose calculation methodology, LME uses XSTERM's XDIS model to calculate post-accident doses to the plant EAB, LPZ, and control room locations. With respect to the information confirmed through RCI-2, as discussed above, the χ/Q input for XSTERM's XDIS model was not always consistent with the consequence level being evaluated (i.e., mean versus 95th). However, the atmospheric dispersion factors were still representative of a 95th percentile value as discussed by RCI-2 above.

The NRC staff reviewed PSAR sections 3.1.5.1 through 3.1.5.5 which describes LME's overall plant risk consequences methodologies. These methodologies describe the NEI 18-04 Task 7b risk metrics for evaluating site boundary risk, early fatality risk, and latent cancer fatality. NRC staff review determined that the proposed methodology is acceptable for the CP application review because the PSAR describes the method that were used to calculate the risk metrics during the CP application and significant margin exists between the preliminary results and the cumulative risk metrics.

In addition, the applicant identified that in a future OL application calculations will be performed to show organ-specific dose calculations for early fatality risk assessment, population-weighted RI over site-specific population distributions, and application of updated risk coefficients consistent with NUREG-1530 or the most current approved-NRC guidance at the time of the OL application submittal. Updating the methodology to calculate the organ-specific doses for the OL application will ensure that the final cumulative risks are calculated in a manner consistent with methods that have been shown to yield appropriate results for these metrics.

The NRC staff reviewed PSAR section 3.10 which describes the non-core related sources of radiological dose. The dose analysis described in this section uses a separate dispersion factor of 5.80×10^{-4} s/m^3 since the dose receptor is located 300-meters away and uses the same breathing rate factor and dose conversion factors used for the DBA analyses. The NRC staff determined that since the same methodology is used to determine the 300-meter dispersion factor as used in the development of the 400-meter dispersion factor reflected in the

atmospheric dose and dispersion methodology TR, this value is acceptable. In addition, PSAR section 3.10 provides reference to the individual source term assumptions used for each analysis.

The NRC staff reviewed PSAR section 3.2.2.2, which states that control room doses for all AOOs and DBEs are less than 5 rem (50 mSv) over 30 days, as summarized in PSAR table 3.2-3. The NRC staff finds the highest dose to the control room for all AOOs and DBEs to be 2.5 rem. PSAR tables 3.2-4 and 3.2-5 provide a mapping of DBEs to DBAs and provide the dose consequences of DBAs in comparison to the 25-rem (250 mSv) criterion for doses at the EAB and outer boundary of the LPZ from 10 CFR 50.34(a)(1)(ii)(D). The NRC staff review confirmed that PSAR section 3.2.3 addresses conformance with offsite dose limits in stating that no BDBE or DBA results in a dose greater than 25 rem. The NRC staff finds the highest 30-day dose at the EAB and outer boundary of the LPZ to be 14.4 rem. Based on the information reviewed during the audit and supplemented with the atmospheric dispersion and dose calculation methodology, the NRC staff determined that the applicant followed an acceptable method for calculation of dose because LME leveraged inputs for source term, dispersion, and breathing rate consistent with the methodology IBR.

In addition, the NRC staff reviewed the non-core related dose analysis reported in PSAR table 3.10-8 and confirmed during audit that the assumptions used to determine the calculated dose consequences are reasonable because the source term represents the events described and the dispersion factor used for non-core sources reflects a closer value needed for the events. During the audit review, the NRC staff confirmed through confirmatory calculations the results of the non-core related doses analysis provided by LME.

Based on its review as described above, the NRC staff determined that the methods to evaluate the radiological consequences of DBAs and other LBEs provides results that are representative of the potential radiological consequences of events at LMGS. In addition, the NRC staff determined that the applicant appropriately described additional consequence methods to address NEI 18-04 Task 7b risk metrics.

3.1.5.2 Conclusion

Based on its review of the atmospheric dispersion and dose calculation methodology described in PSAR section 3.1.5 as discussed above, the NRC staff determined LME meets the regulatory requirements identified in 10 CFR 50.34(a)(1) and (a)(4). The NRC staff's determination of acceptability is based on the applicant's implementation of the referenced NRC-approved TR for atmospheric dispersion and dose calculation analyses along with the associated limitation and conditions.

3.1.6 Core Design Methods and Analyses

PSAR section 3.1.6 provides an overview of the core design and methods for the Xe-100 reactor. The discussion focuses on the determination of important core parameters, such as reactivity coefficients, temperature distribution, power distribution, and shutdown margin, but also includes nuclear design related topics such as neutron flux distributions.

The PSAR incorporates by reference the LTR 006889, "Xe-100 Licensing Topical Report Reactor Core Design Methods and Analysis," Revision 2 (ML25262A123) (hereafter referred to

as the CDMA). The CDMA provides details on the underlying theories, methods, and codes used to evaluate the Xe-100 reactor core.

Applicable regulatory requirements for the evaluation of the core design and methods include:

- 10 CFR 50.34(a)(1);
- 10 CFR 50.34(a)(3)(ii);
- 10 CFR 50.34(a)(3)(iii); and
- 10 CFR 50.34(a)(4).

The following PDCs apply to the core design and methods:

- PDC 10, “Reactor design”
- PDC-RFDC 11, “Reactor inherent protection”
- PDC-RFDC 16, “Functional containment design”
- PDC 20, “Protection system functions”
- PDC-RFDC 26, “Reactivity control systems”
- PDC 28, “Reactivity limits”

The content-of-application guidance in RG 1.253, which endorses NEI 21-07, provides limited guidance on information related to core design. RG 1.70 and RG 1.206 provide guidance to LWR OL and combined license (COL) applicants on nuclear design content of application, while NUREG-0800, section 4.3, Revision 3 (ML070740003), provides licensing review guidance to the NRC staff. This guidance is not directly applicable, since Xe-100 is not an LWR, but the NRC staff finds that NUREG-0800, section 4.3 provides a useful framework for conducting the review.

Similarly, NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors Standard Review Plan and Acceptance Criteria” (ML042430048), provides guidance for reviewing the nuclear design of non-power reactors. This guidance is also not fully applicable because Xe-100 is a power reactor. However, similar to NUREG-0800, the NRC staff used it to inform their review.

3.1.6.1 Technical Evaluation

The NRC staff reviewed the analytical methods for core design and methods discussed in the CDMA to ensure that they are appropriate for demonstrating that the core design supports the applicable PDCs. Specifically, the NRC staff reviewed aspects of nuclear design such as power distribution, reactivity coefficients, and reactor kinetic parameters, source term and consequence analysis as well as use of safety analyses methodologies presented in the CDMA: The acceptability of the analyses and methods to meet the associated PDCs is discussed in the NRC staff’s SE for the CDMA, and the NRC staff’s determination remains applicable to this CP review.

The NRC staff imposed one limitation and one condition on use of the CDMA discussed in the SE. The NRC staff finds that PSAR table 1.4-12 addresses L&Cs for the CDMA. The NRC staff finds, consistent with limitation 1, that the CDMA is being used to support the preliminary safety analysis performed in the PSAR, and is consistent with condition 1, that the regulatory basis supporting the methods described in the CDMA are consistent with the PSAR; and therefore, is acceptable.

Analytical Methods

The analytical methods used for core design and analyses for the Xe-100 reactor are described in detail in section 3 of the CDMA. This section describes the codes used for modeling the core along with general overview of the code system. The results from the core physics methods are discussed in section 4 of the CDMA. Sections 5 and 6 of the LTR describe the plans for code validation as well as V&V of the methodology.

Section 3 of the CDMA describes the modeling techniques and capabilities of the VSOP code used for core design of Xe-100 reactor. This includes details on nuclear cross-section processing, resonance integrals, neutron escape probabilities, material representations within models, and neutron diffusion calculations. The operational aspects such as thermal hydraulic feedback, fuel burnup, and fuel management during reactor operation are also discussed. This section further provides details regarding the use of the STAR-CCM+ code to simulate pebble movements through the Xe-100 reactor core and the generation of appropriate spatial mesh and pebble flow characteristics to provide to the VSOP code. The input generated by STAR-CCM+ to provide to the VSOP code allows tracking and simulation of fuel utilization and isotopics throughout reactor core operations.

The acceptability of core modeling methodology which determines that the calculational approach is adequate to simulate the described Xe-100 reactor core conditions and generate appropriate inputs for use in safety analysis methods is documented in the associated NRC staff's SE of the TR. The NRC staff noted in the CDMA SE that stability of the Xe-100 reactor from operational phenomena, such as Xenon oscillations, was not evaluated in the TR and is expected to be analyzed as part of the final reactor core design and analysis during a future OL application stage. The NRC staff confirmed in a regulatory audit that the preliminary design calculations show that reactor stability for the Xe-100 reactor is maintained through inherent reactivity feedback as well as engineered response to operational phenomena, such as Xenon transients.

The NRC staff finds that the analytical methods used for the core design and analysis of the Xe-100 presented in the CDMA TR acceptable since the TR is IBR and LME has addressed the associated L&Cs as discussed above.

Core Design Results

Section 4 of the CDMA provides results from reactor core physics calculations for key phenomena including reactivity coefficients, Xenon feedback, shutdown reactivity worth, axial and radial reactor core power distributions as well as a representative six-group delayed neutron data. This section also provides results for the STAR-CCM+ relative velocity for each of the radial channels which is then used as an input into the VSOP code. The NRC staff noted in the SE for the CDMA that while the results presented demonstrate the capability of the codes to model important physical characteristics of the Xe-100 reactor to support the downstream safety

analysis, they are not intended to address any specific safety analysis needs for licensing a Xe-100 reactor.

PSAR section 3.1.6 presents results for key core design and neutronics phenomena for the equilibrium core. Specifically:

- PSAR figures 3.1-2 and 3.1-3 present the axial fast and thermal flux distributions for the core at three radial positions along the core centerline, inner side reflector (graphite) and outer edge of the outer reflector (graphite).

- PSAR figure 3.1-4 presents fast and thermal radial flux distributions at the axial location of the greatest flux.

- PSAR figure 3.1-5 presents equilibrium axial power distributions for the core at three radial positions along the core centerline, inner side reflector (graphite) and outer edge of the outer reflector (graphite) while figure 3.1-6 shows radial power distribution at a level of 479 centimeters (cm) below the top core barrel.

- PSAR figures 3.1-7 and 3.1-8 present axial and radial temperature distributions of the core as a function of depth below top reflector and radial width respectively. The results show that axial pebble temperature gradient in the core is steepest towards the top portion of the core where core power and flux are at their maximum.

- PSAR figure 3.1-9 shows the axial helium temperature profile in the equilibrium core at select radial positions. The PSAR notes that helium temperatures drop at the bottom periphery of the core due to bottom cone and single-stream bypass model leading to local cooling on the hot helium passing through the core.

- PSAR figure 3.1-10 presents integral rod worth for Reactivity Control System (RCS) and Reserve Shutdown System (RSS) rods. During normal operation, the nine RCS rods are inserted at depths of 170 cm below the bottom of the top reflector. The figure shows total core excess reactivity with rods fully withdrawn is 1556 pcm. The figure shows that RSS rods insert very little negative reactivity until they exceed the depth of the RCS rods. The rod worths for equilibrium core and total shutdown margin at hot and cold conditions are presented in PSAR table 3.1-2. The table shows shutdown margin with one rod assumed stuck out for each of the rod banks.

- Figure 3.1-11 shows temperature coefficients of reactivity for the fuel, moderator and reflector regions. The figure shows strong negative overall temperature coefficient to ensure inherent protection from reactivity increases. The figure shows that Graphite reflector coefficient is positive due to reduced neutron capture cross-section at higher temperatures.

In the regulatory audit, the NRC staff confirmed that the input parameters and design characteristics used to generate the results presented in the PSAR are based on the analytical methods described in the CDMA. Hence, the NRC staff finds that the results presented in the PSAR, along with the description of the core design methods, are acceptable and adequately support the relevant PDCs described in the SE.

The PSAR states that LME will conduct uncertainty analysis of the kinetic parameters, reactivity coefficients and other core design parameters along with appropriate sensitivity studies on these parameters during the development of the FSAR at OL application.

Validation and Verification

Section 5 of the CDMA provides an overview of the validation plans for VSOP and STAR-CCM+, including plans to include comparisons to experimental datasets. The LTR lists three experimental data sets to be relied upon to support validation of the VSOP code. The parameters of the experimental datasets available for comparison to support code validation include criticality, temperature coefficients, control rod worths, steady-state operations, power ascension operations, kinetics parameters, reflector worth, pebble packing fraction, and mixed cores of graphite moderator, poison, and fuel pebbles. Similarly, code-to-code benchmarking is identified for inclusion in the validation plan which will be performed to compare the VSOP against higher-order codes.

In the SE for the CDMA, the NRC staff found that the experimental tests identified for comparison to be similar in design to the described Xe-100 and, therefore, are expected to provide useful comparisons to support the validation of the VSOP methods described in the TR. Additionally, the planned validation cases and associated parameters of interest for comparison, cover the range of parameters that VSOP provides as input to the overall transient safety analysis EM.

The LTR states that the validation basis for STAR-CCM+ will rely on the comparison to the ANABEK experiment, which focused on characterizing pebble flow behaviors at scale. The PSAR states that the ANABEK experiment has geometric parameters that are similar to the Xe 100 design with respect to pebble flow because the ANABEK experiment and Xe-100 design have similar values for:

- the ratio of the core height to the diameter,
- the ratio of the core diameter to the pebble diameter,
- the ratio of the defuel chute diameter to the pebble diameter, and
- the outlet funnel’s conical angle to the horizontal plane.

This comparison will focus on the non-dimensional retention time ratio that describes the pebbles’ relative in-core time through the centerline of the reactor core versus at the wall, and the residence time ratio, which describes a measure of pebble residence time normalized to the recirculation frequency.

In the SE for the CDMA, the NRC staff found that the STAR-CCM+ validation approach to be acceptable since it is based on comparison to experimental evidence and is an approach that has been demonstrated successfully in the other similar works referenced in the CDMA.

Section 6 of the CDMA lists planned verification activities which includes examining the validation exercise and associated reports to confirm the validation plan. The verification will also ensure that the benchmark data used in the validation exercise is suitably qualified and any bias and variations in the data are quantified for code accuracy. The LTR self-imposed a

limitation which states that the VSOP and STAR-CCM+ codes cannot be used to support a FSAR until the V&V of the codes are complete and have been approved by the NRC.

In the SE for the CDMA, the NRC staff concluded that the verification plan appropriately described the objective of the V&V including determining the adequacy of the validation methods, identifying existing test databases and performing evaluations to establish model adequacy.

The NRC staff finds the validation approach and verification plan described in the CDMA TR to be acceptable for a CP since the TR is IBR and LME has addressed the associated L&Cs.

3.1.6.2 Conclusion

Based on the discussion above, the NRC staff determined that the core design methods and analyses provided in PSAR section 3.1.6, along with the CDMA LTR IBR, is adequate to support preliminary analyses demonstrating the Xe-100 reactor is consistent with PDCs 10, 20 and 28 and PDC RFDCs 11, 16, and 26. The PSAR provides sufficient results for the NRC staff to confirm that the preliminary nuclear design supports these PDCs. The NRC staff determined that the preliminary evaluation included in the CDMA provides adequate preliminary analysis and evaluation of the design per 10 CFR 50.34(a)(4), and the core design information provided in the PSAR is sufficient for the NRC staff to determine that the applicant adequately described the design bases and the relationship of the design bases to the PDCs per 10 CFR 50.34(a)(3)(ii). The areas that require further development prior to the issuance of an OL, per 10 CFR 50.35, are discussed above.

3.2 Licensing Basis Event Summary and Evaluation Methodology

PSAR section 3.2 summarizes the LBEs identified for LMGS and includes a general description of the LBE evaluation process. Methods for evaluating LBEs are found in PSAR section 3.1. The LBE selection methodology is described in PSAR section 3.1.1.4. PSAR sections 3.1.3 through 3.1.6 describe the analytical methods used to evaluate AOOs, DBEs, and BDBEs and PSAR sections 3.3 through 3.5 describe the evaluation of the AOOs, DBEs, and BDBEs. PSAR section 3.1.2 describes the analytical methods used to evaluate DBAs and PSAR section 3.6 describes the evaluation of the DBAs.

PSAR section 3.2 summarizes the AOOs, DBEs, and BDBEs, including their risk significance in a series of tables and figures. The dose requirements listed in 10 CFR 50.34, “Contents of applications; technical information,” are used to classify a BDBE as high-consequence. PSAR section 3.2, table 3.2-4 summarizes non-DBAs LBEs (i.e., AOOs, DBEs, and BDBEs) and their connections to DBAs.

The applicable regulatory requirements for the evaluation of the LBE summary are as follows:

- 10 CFR 50.34(a)(1) and (4) of 10 CFR; and
- 10 CFR 50.35.

The applicable guidance for the evaluation of the LBE summary is as follows:

- RG 1.233, Revision 0, which endorses NEI 18-04, Revision 1, with clarifications;

- RG 1.253, Revision 0, which endorses NEI 21-07, Revision 1, with clarifications and additions.

3.2.1 Technical Evaluation

PSAR chapter 3, states, “Chapter 3 combines the content described in sections 2 and 3 of NEI 21-07.” Accordingly, the NRC staff evaluated PSAR section 3.2, “Licensing Basis Event Summary and Evaluation Methodology,” against the criteria provided in NEI 21-07, section C.3.2, “LBE Summary,” as endorsed by RG 1.253. Specifically, sections 3.2.1.1 and 3.2.1.2 of this SE evaluates PSAR section 3.2 against the six criteria from NEI 21-07, section C.3.2.1, “Summary Evaluation of AOOs, DBEs, and BDBEs,” and the two criteria from NEI 21-07, section C3.2.2, “Summary Evaluation of DBAs.”

Additionally, the NRC staff evaluated the information supporting PDC owner controlled design criteria (OCDC) 19 in section 3.2.1.3 of this SE and the information relevant to dose requirements under 10 CFR 50.34(a)(1)(ii)(D) in section 3.2.1.4 of this SE.

3.2.1.1 Summary Evaluation of AOOs, DBEs, and BDBEs

As described in RG 1.253, NEI 21-07, section C.3 provides an acceptable method for developing information in chapter 3 of the safety analysis report related to the summary of LBEs (AOOs, DBEs, BDBEs, and DBAs). NEI 21-07, section 3.2.1 states that the summary of the evaluation of LBEs derived from the PRA should include:

- Tables with brief word descriptions of the AOOs, DBEs, and BDBEs.
- Identification of the radionuclide sources associated with each of the LBEs.
- A plot of the frequencies, consequences, and uncertainties of these LBEs with comparison to the NEI 18-04 F-C Target in figure 3-1 of NEI 18-04.
- Identification of all risk-significant LBEs as defined in NEI 18-04.
- Identification of any high-consequence BDBEs as defined in NEI 18-04, (i.e., those BDBEs with EAB doses greater than 25 rem.)
- Definition of the reactor-specific safe, stable end states, described previously and used to establish the SC for the safety functions modeled in the PRA, referred to in NEI 18-04 as PSFs and reflected in the LBE descriptions.

The NRC staff’s evaluation PSAR section 3.2 and LME’s implementation of the guidance in NEI 21-07, section 3.2.1 and RG 1.253 is provided in subsections 3.2.1.1.1 through 3.2.1.1.7 of this SE.

3.2.1.1.1 Tables with brief word descriptions of the AOOs, DBEs, and BDBEs

NEI 21-07, section 3.2.1, states: “The word descriptions of the LBE should be in sufficient detail to indicate the PRA Safety Functions involved in the prevention and mitigation of the LBEs.”

The NRC staff reviewed PSAR table 3.2-2, "Licensing Basis Events (Sorted by Frequency)," which summarizes the non-DBA LBEs (AOOs, DBEs, and BDBEs). PSAR table 3.2-2 provides the ES designation, LBE categorization type, IE, and describes the plant response for the non-DBA LBEs. The plant response descriptions include the PSFs involved in the prevention and mitigation of the LBEs (e.g., reactor trip (RT), power runback, heat removal, steam isolation). PSAR table 3.2-2 does not identify specific actuation triggers or logic used to initiate prevention and mitigation functions and finds this to be acceptable for the summary descriptions at the CP stage because this level of detail is not needed to identify PSFs but will be needed at the OL application stage to confirm that all SSCs necessary to perform the PSF have been identified and provided with the appropriate classification and special treatment. PSAR table 3.2-2 also provides the mean, 5th percentile, and 95th percentile frequencies, the mean and 95th percentile consequences, and whether the LBE is risk significant. Based on the information provided in PSAR table 3.2-2, the NRC staff finds that the applicant provided reasonable summaries of the AOOs, DBEs, and BDBEs because the information provided is consistent with NEI 21-07, section 3.2.1, as endorsed by RG 1.253.

3.2.1.1.2 Identification of the radionuclide sources associated with each of the LBEs

The NRC staff reviewed PSAR section 3.2 which states, in part, that, "the scope of the PRA is limited to radionuclides within the [Helium Pressure Boundary], full-power operation, and internal events." The NRC staff's evaluation of the PRA scope, provided in section 3.1.1 of this SE, finds that the PRA scope is reasonable to support a CP application. Based on the information provided in PSAR sections 3.1.1.2 and 3.2, the NRC staff finds that the applicant identified the radionuclide sources associated with each LBE because: (1) the scope of radionuclides considered in the PRA is clearly stated, and (2) the scope of the PRA is reasonable to support the CP application as discussed in section 3.1.1 of this SE.

3.2.1.1.3 A plot of the frequencies, consequences, and uncertainties of these LBEs with comparison to the NEI 18-04 Frequency-Consequence Target in figure 3-1 of NEI 18-04

The NRC staff reviewed PSAR section 3.2.2.1, "Exclusion Area Boundary (EAB) Results," which describes the following figures:

- PSAR figure 3.2-2 compares LBEs with mean consequence at the EAB greater than $1x10^{-4}$ rem (0.1 mrem) against the F-C target.
- PSAR figure 3.2-3 compares risk-significant LBEs against the F-C target.
- PSAR figure 3.2-4 compares the AOOs, DBEs, and BDBEs evaluated in PSAR sections 3.3 through 3.5 against the F-C target.

NEI 21-07, section 3.2.1, states:

> The plots of the LBE frequencies and consequences should be made with points corresponding to the mean estimates of frequency and consequence with uncertainty bars indicating the 5th and 95th percentiles of the quantified uncertainty distributions in both frequency and dose for each LBE.

PSAR figures 3.2-2 through 3.2-4 provide frequencies and consequences with uncertainties at the 95th percentile but deviate from NEI 21-07, as endorsed by RG 1.253, by not including uncertainties at the 5th percentile for consequences. PSAR table 3.2-2 provides the mean, 5th percentile, and 95th percentile frequencies and the mean and 95th consequences for each LBE. PSAR section 3.2.2.1 states: "Because of the large uncertainties in the preliminary dose calculations, the 5th-percentile consequences are not shown, [in PSAR table 3.2-2] but will be included in the [OL application]." The NRC staff finds this deviation from NEI 21-07 to be acceptable at the CP stage because the LBE 5th percentile consequence does not change the categorization or risk significance of the LBEs. The NRC staff will review the 5th percentile consequence for the LBEs in the OL application stage.

NEI 21-07, section 3.2.1, also states, "LBEs with no release and hence no dose should be plotted on the Y-axis." PSAR figure 3.2-2 shows LBEs with mean consequences at the EAB greater than $1x10^{-4}$ rem (0.001 mSv). The applicant stated in PSAR section 3.2.2.1 that to reduce crowding on figure 3.2-2 lower dose LBEs are not represented. The NRC staff finds this deviation from NEI 21-07 to be acceptable because PSAR table 3.2-2 provides the mean frequency and consequence for each LBE. For the CP application stage, LME has chosen not to use ES families. The NRC staff notes that when ES families are not used, the plotting of all low or no consequence LBEs makes the plot difficult to read and therefore, providing the mean frequency and consequence information for the LBEs in tabular form is acceptable.

Consistent with NEI 18-04, the mean values of the frequencies shown in PSAR figure 3.2-2 and listed in PSAR table 3.2-2 are used to classify the LBEs into AOOs, DBEs, and BDBEs. When the uncertainty bands defined by the 5th and 95th percentiles of the frequency estimate straddles a frequency boundary, the LBE is evaluated in both LBE categories. DBE doses are evaluated against the F-C target based on the mean estimates of consequences. PSAR table 3.2-4 provides which DBA is used to evaluate each non-DBA LBE (AOOs, DBEs, and BDBEs). For each AOO with the 5th percentile in the DBE region and for each BDBE with the 95th percentile in the DBE region, a DBA evaluation is either performed for the LBE or a representative DBA evaluation is selected for the LBE. A DBA is identified for each DBE and is then evaluated via a conservative safety analysis using only SR SSCs.

The NRC staff finds the applicant's deviations related to plotting the frequencies, consequences, and uncertainties of the LBEs against the F-C target, in accordance with NEI 21-07, as endorsed by RG 1.253, to be acceptable because: (1) the applicant provided the information in a tabular format and (2) the missing 5th percentile consequence values for the LBEs does not change the categorization or risk significance of the LBEs or SSCs. The NRC staff will review the 5th percentile consequences for the LBEs at the OL application stage. Except for these deviations, the plots of the frequencies, consequences, and uncertainties of LBEs against the F-C target are consistent with NEI 18-04, as endorsed by RG 1.233. Based on the evaluation above, the NRC staff finds that the plotted frequencies, consequences, and uncertainties of the LBEs against the F-C target is acceptable.

3.2.1.1.4 Identification of all risk-significant LBEs as defined in NEI 18-04

NEI 18-04, as endorsed by RG 1.233, provides the following definition for risk-significant LBE:

> An LBE whose frequency and consequence meet a specified risk significance criterion. In the LMP framework, an AOO, DBE, or BDBE is regarded as risk-

significant if the combination of the upper bound (95th percentile) estimates of the frequency and consequence of the LBE are within 1 percent of the F-C Target AND the upper bound 30-day TEDE dose at the EAB exceeds 2.5 mrem (0.025 mSv).

The NRC staff reviewed PSAR figure 3.2-3 and identified the AOOs, DBEs, and BDBEs provided in table 3.2-1 of the SE as risk-significant LBEs. The NRC staff compared the list developed from PSAR figure 3.2-3 to the AOOs, DBEs, and BDBEs provided in PSAR figure 3.2-2 and PSAR table 3.2-2 against the criteria stated in NEI 18-04 task 7c. Based on this comparison, the NRC staff determined that PSAR figure 3.2-3 reasonably identifies all risk-significant AOOs, DBEs, and BDBEs because these are the LBEs that meet the risk significance criteria defined in NEI 18-04, as endorsed by RG 1.233.

Table 3.2-1: Risk-Significant AOOs, DBEs, and BDBEs

AOOs	DBEs	BDBEs
SD-068	SRT-018	SFLLBRB-01
SSGTL-03	SSGTL-14	SFLMBRB-04
MTLHS-01	SD-076	LD-01
MPRBC-08	MD-63	SSLMBRB-01
MTLFW-01	MD-68	SFLLBRB-04
MLOIA-01	MD-73	SSLLBRB-01
MLOIA-06	SFLMBRB-01	SSGTL-25
MPLFW-02		SRT-027
MPLFW-01		STLPF-31
MPRBC-01		SD-131
MPRBC-02		SSGTL-06
MTLHS-06		STLPF-10
MPLFW-08		MTRBC-04
MPLHS-02		SRT-021
MPLHS-01		SSGTL-17
SPLPF-08		SRT-032
SPLPF-02		SRT-006
SPLPF-01		STLPF-25
		MLOIA-05
		SSGTR-06
		STLPF-36

The NRC staff finds that the applicant identified all risk-significant LBEs as defined in NEI 18-04 because: (1) the applicant used an acceptable method to select LBEs for preliminary safety analysis as discussed in section 3.2.1.1.6 of this SE, and (2) PSAR figure 3.2-3 and PSAR table 3.2-2 reasonably identify all risk-significant AOOs, DBEs, and BDBEs.

3.2.1.1.5 Identification of any high-consequence BDBEs as defined in NEI 18-04

NEI 18-04, as endorsed by RG 1.233, provides the following definition for high-consequence BDBEs:

> High consequence BDBEs are those with consequences that exceed 10 CFR 50.34 dose criteria.

The NRC staff reviewed PSAR section 3.2, which states: "There are no high-consequence BDBEs as defined in NEI 21-07, as none result in an EAB dose greater than 25 rem." The NRC staff reviewed PSAR figure 3.2-2 and confirmed that no LBEs exceed the 10 CFR 50.34 dose criteria. Additionally, the NRC staff reviewed PSAR table 3.2-2 and finds that for all LBEs the associated mean and 95th percentile consequences remains below the 10 CFR 50.34 dose criteria. The NRC staff finds reasonable assurance that the LMGS contains no high-consequence BDBEs, as defined in NEI 18-04 as endorsed in RG 1.233, because: (1) the applicant used an acceptable method to select LBEs for preliminary safety analysis and to evaluate AOOs, DBEs, and BDBEs as discussed in sections 3.2.1.1.6 and 3.3 through 3.5 of this SE, and (2) no LBEs were identified that meet the criteria for a high-consequence BDBE.

3.2.1.1.6 Definition of the reactor-specific safe, stable end states, described previously and used to establish the success criteria for the safety functions modeled in the PRA and LBE Selection Methodology

As described in RG 1.253, after identifying the LBEs, chapter 3 of a safety analysis report, should describe the systematic and reproducible process and methodology used to select the LBEs, including the process used to group and condense the ESs into ES families, and the specific analysis and evaluation of the selected LBEs for the proposed design. NEI 21-07 section 3.1 states that safe, stable end states are a key element of the reactor safety case and should be defined in the LBE selection methodology section of the safety analysis report, including the plant parameters used to define the end states. Additionally, NEI 21-07 states that the safe, stable end states, including SC that are needed to achieve them, need to be defined for the specific technology and design. As described in RG 1.253, the analyses in chapter 3 of a safety analysis report following NEI 21-07, are primarily described in terms of ESs consisting of an IE, the plant response to the IE, and a well-defined end state.

The NRC staff reviewed PSAR section 3.1.1.4, "LBE Selection Methodology," which states that non-DBA LBEs are identified by their probability of occurrence, as determined by the LMGS PRA model, and provides a range of probabilities of occurrence for each LBE category. PSAR section 3.1.1.4 also states that DBAs are derived from DBEs. PSAR section 3.1.1.5, "Safe and Stable Endpoint," describes the definition of a safe, stable end state specific to the Xe-100 design.

To further understand the description in PSAR sections 3.1.1.4 and 3.1.1.5, the NRC staff conducted an audit of the following information:

- Document ID number 000897, "Probabilistic Risk Assessment: Initiating Event Analysis – Preliminary Scoping," Revision 4
- Document ID number 001994, "Xe-100 Probabilistic Risk Assessment: Success Criteria Development," Revision 3

- Document ID number 002414, “Probabilistic Risk Assessment: Event Sequence Analysis – Preliminary Scoping,” Revision 4

- Document ID number 005147, “Probabilistic Risk Assessment: Event Sequence Quantification Preliminary Scoping Analysis Report,” Revision 6

- Document ID number 006869, “Probabilistic Risk Assessment: Mechanistic Source Term Analysis – Preliminary Scoping,” Revision 3

The audit focused on confirming that LME’s LBE selection process aligns with the LBE selection process in NEI 18-04, as endorsed by RG 1.233. The NRC staff assessed the LBE selection process and specifically concentrated on confirming event category definitions and any associated demarcations used in the process to confirm that the methodology plotted ESs on the F-C targets, assessed margins based on event frequency, and estimated 30-day dose at the EAB. The audit focused on whether the LBE process facilitates the determination of risk-significant LBEs and SSCs and the evaluation of DID adequacy. Overall, the NRC staff’s audit focused on ensuring that the LBE selection process appropriately reflects the LMGS design and aligns with NEI 18-04 for the current stage of design maturity.

The NRC staff reviewed PSAR sections 3.1.1.4, 3.1.1.5, and 3.2, and the overall LBE type classifications documented in PSAR chapter 3 against the guidance in RG 1.233 and NEI 18-04, including the technical basis for the LBE selection and categorization methodology. The NRC staff’s review considered the multiple tables and figures in PSAR section 3.2 categorizing and describing the LBEs. The LBEs are derived from the ESs developed in the site and design-specific PRA and are categorized according to their frequency into one of the following three groups:

- AOOs
- DBEs
- BDBEs

DBAs are also considered LBEs and are derived from the set of DBEs, AOOs and BDBEs with frequency bands extending into the DBE region by deterministically assuming that only SR SSCs are available to respond to the event. Section 3.6 of this SE documents the NRC staff’s evaluation of the DBA selection and analysis. The NRC staff confirmed that the initial LBE selection process is consistent with NEI 18-04, in that an initial set of LBEs was proposed based on a review of the plant design and iterated with the PRA to arrive at a final set of LBEs. The LBEs are categorized and analyzed outside of the PRA to determine their radiological consequences. The NRC staff’s evaluation of the LBEs in the AOO, DBE, and BDBE categories is in sections 3.3 through 3.6 of this SE. The NRC staff’s evaluation of the methodologies and analyses used to analyze the AOOs, DBEs, BDBEs, and DBAs is in sections 3.1.2 through 3.1.6 of this SE. Section 3.1.1 of this SE documents the NRC staff’s evaluation of the PRA, which includes the IEs, SC, ESs, and ESQ.

Consistent with the guidance in section 3.1 of NEI 21-07, the LBE analysis end states are detailed in PSAR section 3.1.1.5. The safe, stable end states consist of the following parameters grouped into six criteria:

1. Reactor power is in thermal equilibrium with heat removal via SG, the reactor may be operating at full power, reduced power, or shutdown.

2. Fuel and reflector temperatures are constant or decreasing and within design limits.

3. Primary helium temperature, pressure, radioactivity, and impurity levels are maintained within design limits and either constant or decreasing.

4. Reactor vessel, SG, and HPB component temperatures are either constant or decreasing and within design limits.

5. Control rods and shutdown rods are in fixed positions and may be withdrawn during reactor power operation or inserted if the reactor is shutdown.

6. Reactor vessel supports, reactor vessel citadel walls, and SG citadel concrete temperatures are within design limits.

All six criteria must be met for each reactor module to be considered in a safe, stable end state. For the current stage of design maturity, the MST analysis identified, and the PRA model included thirty-two different unique release end states. The NRC staff's evaluation of the MST methodology is documented in section 3.1.4 of this SE. The NRC staff reviewed the end states referenced in the PRA model and included in the SC, ES, and ESQ, technical elements in the ASME/ANS non-LWR PRA standard. The NRC staff's evaluation of these PRA elements is in section 3.1.1 of this SE. LME did not include the design limits in the PSAR. The NRC staff determined that this is acceptable at the CP stage because the preliminary analysis presented in the PSAR for LBEs demonstrates margin to the F-C target in NEI 18-04. The NRC staff will review the design limits and their associated analyses at the OL application stage. Overall, the NRC staff finds that PSAR section 3.1.1.5 describes the safe, stable end state in a level of detail suitable for the preliminary plant design.

As discussed above, the NRC staff reviewed the LBE selection methodology and the safe, stable end state described in PSAR sections 3.1.1.4 and 3.1.1.5, along with the LBE information in PSAR section 3.2 and determined that the LBE selection process is consistent with NEI 18-04 and the guidance in RG 1.233, providing an acceptable method for identifying and categorizing events. In addition, PSAR sections 3.1.1.4 and 3.1.1.5 are consistent with NEI 21-07 and the guidance in RG 1.253, describing the information related to the LBE selection process and safe, stable end state.

3.2.1.1.7 Identification of Bounding or Representative LBEs

The applicant divided each set of AOOs, DBEs, BDBEs, and DBAs into the following types of transients:

- Reactor trip (RT)
- Turbine trip (TT)
- Main steam line break (MSLB)
- Main feedwater line break (MFLB)
- Control rod group withdrawal (CRW)
- Small HPB depressurization (SD)
- Medium HPB depressurization (MD)
- Large HPB depressurization (LD)

- Loss of primary flow (LPF)
- Loss of steam generator feedwater (LFW)
- Steam generator tube leak (SGTL)
- Steam generator tube rupture (SGTR)
- Loss of reactor building cooling water (RBC)
- Loss of instrument air (LOIA)
- Loss of condenser heat sink (LHS)
- Loss of offsite power (LOOP)

The LBEs in these transient categories are discussed in sections 3.3 through 3.6 of this SE. The NRC staff reviewed PSAR section 3.2 against the guidance in RG 1.253 and NEI 21-07 for summarizing the LBE information described in the PSAR. The NRC staff's review considered the multiple tables and figures in the PSAR chapter 3 categorizing the LBEs and plotting the LBE frequencies and radiological consequences in comparison to the F-C chart in NEI 18-04, figure 3-1. PSAR table 3.2-2 provides numerical values for the data represented in PSAR figures 3.2-2 through 3.2-4. The tables in PSAR chapter 3 provide a description of the plant response for each LBE, the LBE category type and initiator, risk significance, frequencies, and radiological dose consequences. The PSAR figures compare the LBE results to the F-C target in NEI 18-04, figure 3-1.

In addition to reviewing the LBEs based on their categorization (i.e., AOO, DBE, BDBE, and DBA), the NRC staff reviewed the LBEs and the credit attributed to each preventative and mitigative function within the safety analysis. The applicant did not develop unique source terms or perform consequence modeling for all ESs identified in the PSAR. For some ESs, the applicant chose analyses for other LBEs representative or bounding of similar events. The NRC staff determined that this approach is acceptable at the CP stage when design information is limited, but additional analyses will be needed at the OL application stage to appropriately capture differences in the mechanisms for release and the effectiveness of mitigative controls for the mechanistic source terms. Sections 3.1.4 and 3.1.5 of this SE describe the NRC staff's evaluation of the mechanistic source terms and release mechanisms, respectively.

The NRC staff reviewed the nomenclature selected by the applicant to name LBE sequences. Table 3.2-2 of this SE contains the IE root identifiers for the LBEs included in the PSAR and grouped by the type of transient. In reviewing the LBE analyses throughout chapter 3, the NRC staff identified that IEs have the same root identifier, with the success case (i.e., all controls functioning) identified with a -1 extension and events representing failure of preventative or mitigative control being identified with a higher numerical extension (e.g., -2, -3, -4) with higher numbers generally representing more failures. The NRC staff also identified that the IEs have a pre-designator that describes whether a single module or multiple modules are affected and, for loss of flow initiators, whether there is a total or partial loss of flow.

Each IE has multiple associated LBEs, which reflect different PRA safety function failures. The NRC staff used the associated LBEs for each IE to assess the preventative and mitigative effects of the PSFs and to identify the expected SSC safety classification based on those effects. Section 5 of this SE describes the NRC staff's evaluation of the preventative and mitigative effects and classification of SSCs.

The NRC staff reviewed PSAR section 3.2, which states that the applicant assigned bounding radiological dose consequences to each LBE based on the LBE's event progression, release

category, IE, and the plant's end state. The LBE sequences were grouped according to their progression path and long-term cooling method at the end state. The applicant determined that the offsite radiological dose released for each LBE is driven by the LBE's end state and accordingly did not use the LBE initiator (e.g., LOOP, LOIA, MD) to choose the representative or bounding dose scenario, but instead used the end state (e.g., pressurized RCCS cooling, depressurized RCCS cooling etc.). However, because of the preliminary maturity of the design at the CP application stage, not all end states are analyzed; therefore, the applicant assigned the largest analyzed radiological dose, the seismic-B scenario, to bound the consequences for the LBE sequences that end in degraded RCCS or included a failure to trip in the ES.

The NRC staff evaluated the information in PSAR table 3.2-2 to ensure that the ESs described have bounding dose consequences assigned. The NRC staff audited the event descriptions to confirm that they are consistent with the information contained in PSAR table 3.2-2, along with the assumed end state information for the events provided in PSAR table 3.2-2. Based on a sampling of events, the NRC staff determined that the event descriptions along with the expected end states seem to reasonably bound the dose consequences expected for the ESs.

In addition to the six criteria identified in NEI 21-07, as endorsed by RG 1.253, the NRC staff reviewed the information in PSAR table 3.2-2 to identify which AOOs, DBEs, BDBEs, and DBAs should be described in greater detail in PSAR section 3.3, 3.4, 3.5, and 3.6, respectively. NEI 21-07 includes guidance on the scope and level of detail that should be used to describe the AOOs, DBEs, BDBEs, and DBAs. Because LME did not perform the LBE family grouping and limited unique consequence evaluations were available to support the PSAR, the PSAR content in section 3.3-3.6 does not align with the content guidance in NEI 21-07. LME describes this in section PSAR section 3.2.4. The NRC staff performed a review comparing all the LBEs identified in PSAR table 3.2-2 to those described in PSAR sections 3.3 through 3.5 to identify the scope of the deviations and determine which LBEs may have unique ES descriptions that are not represented in the PSAR. Table 3.2-3 of this SE lists the AOOs, DBEs and BDBEs evaluated in PSAR sections 3.3 through 3.5.

The NRC staff finds the twenty-five IE types listed in table 3.2-4 of this SE, which represents 157 LBEs, have no LBEs explicitly represented in PSAR sections 3.3 through 3.5. However, as discussed in PSAR section 3.2.4, many of these IE types are substantially similar to those LBEs that are explicitly represented in the PSAR. The mapping for what IE types are substantially similar is provided in PSAR section 3.2.4. Of the twenty-five IE types, the NRC staff determined that nine were not similar to those represented in PSAR sections 3.3 through 3.5. Therefore, LME added PSAR table 3.2-7, which provides ES descriptions for the nine IE types that aren't explicitly represented and explains the unique aspects to their ES progression. The NRC staff finds this level of information to be sufficient for the CP stage based on summary descriptions being provided for all events in PSAR table 3.2-2, each event type being represented in PSAR sections 3.3 through 3.6 or being described in PSAR section 3.2.4 in sufficient detail for the NRC staff to understand the event progression and confirm that the preliminary design is consistent with the events described.

Table 3.2-2: Initiating Events Reflected in the PSAR

Transient Type	Number of non-DBA LBEs	Initiating Event Identification	Initiating Event Description
Reactor trip	16	SRT	Single module reactor trip
	10	MDRT	Multi-module double reactor trip
	5	MPRT	Multi-module plant-wide reactor trip
Turbine Trip	8	MTT	Multi-module turbine trip
Control rod group withdrawal	13	SCRW	Single module control rod group withdrawal
	4	MCRW	Multi-module control rod group withdrawal
Main steam line break	5	MSLB	Main steam line break
	9	SSLBNI	Single module main steam line break in NIAS
	5	SSLSBRB	Single module main steam line small break in RB
	2	SSLMBRB	Single module main steam line medium break in RB
	2	SSLLBRB	Single module main steam line large break in RB
Main steam header imbalance	6	MSTIS	Single valve steam header imbalance
	2	MSTID	Double valve steam header imbalance
	2	MSTIT	Triple valve steam header imbalance
Main feedwater line break	4	MFLB	Main feedwater line break
	7	SFLDBNI	Single module feedwater line break downstream flow element in NIAS
	6	SFLUBNI	Single module feedwater line break upstream flow element in NIAS
	3	SFLSBRB	Single module feedwater line small break in RB
	2	SFLMBRB	Single module feedwater line medium break in RB with induced medium HPB depressurization only on module #1
	2	SFLLBRB	Single module feedwater line large break in RB with induced large HPB depressurization only on module #1
Main feedwater header imbalance	6	MFWIS	Single valve feedwater header imbalance

Transient Type	Number of non-DBA LBEs	Initiating Event Identification	Initiating Event Description
	3	MFWID	Double valve feedwater header imbalance
	2	MFWIT	Triple valve feedwater header imbalance
Small HPB depressurization	14	SD	Small HPB depressurization
Medium HPB depressurization	9	MD	Medium HPB depressurization
Large HPB depressurization	1	LD	Large HPB depressurization
Loss of primary flow	17	SPLPF	Single module partial loss of primary flow
	12	MPLPF	Multi-module partial loss of primary flow
	11	STLPF	Single module total loss of primary flow
	3	MTLPF	Multi-modules total loss of primary flow
Loss of steam generator feedwater flow	8	MPLFW	Multi-module partial loss of steam generator feedwater flow
	5	MTLFW	Multi-module total loss of steam generator feedwater flow
Steam generator tube leak	22	SSGTL	Single module steam generator tube leak
Steam generator tube rupture	14	SSGTR	Single module steam generator tube rupture
Loss of reactor building cooling water	8	MPRBC	Multi-module partial loss of reactor building cooling water
	5	MTRBC	Multi-module total loss of reactor building cooling water
Loss of Instrument Air	8	MLOIA	Multi-module loss of instrument air
Loss of condenser Heat Sink	8	MPLHS	Multi-module partial loss of condenser heat sink
	10	MTLHS	Multi-module total loss of condenser heat sink
Loss of offsite power	10	MLOOP	Multi-module LOOP

Table 3.2-3: AOOs, DBEs, and BDBEs Presented in PSAR

AOO	Initiator	PSAR Section
MTT-01, MTT-02	Turbine Trip	3.3.1
MSLB-01	Main Steam Line Break	3.3.2
SCRW-01	Control Rod Withdrawal	3.3.3
SD-68	Small Helium Pressure Boundary Depressurization	3.3.5
MTLFW-01	Loss of Feedwater	3.3.6
SSGTL-01	Steam Generator Tube Leak	3.3.7
MLOOP-01	Loss of Offsite Power	3.3.8
DBE	**Initiator**	**PSAR Section**
MLOOP-23	Loss of Offsite Power	3.4.1
MTLPF-22	Loss of Primary Flow	3.4.2
MD-68	Medium Helium Pressure Boundary Depressurization	3.4.3
MFLB-01	Main Feed Water Break	3.4.4
SSGTR-14	Steam Generator Tube Rupture	3.4.5
SCRW-13	Control Rod Withdrawal	3.4.6
MTRBC-01	Loss of Reactor Building Cooling Water	3.4.9
BDBE	**Initiator**	**PSAR Section**
SCRW-03	Control Rod Withdrawal	3.5.1
SD-74, SD-131	Small Helium Pressure Boundary Depressurization	3.5.2
MD-65	Medium Helium Pressure Boundary Depressurization	3.5.3
LD-01	Large Helium Pressure Boundary Depressurization	3.5.4
MTLFW-03	Loss of Feedwater	3.5.5
MFLB-02	Feedwater Line Break	3.5.6
MSLB-08	Main Steam Line Break	3.5.7
SSGTL-06, SSGTL-21	Steam Generator Tube Leak	3.5.8
SSGTR-29	Steam Generator Tube Rupture	3.5.9
STLPF-25	Loss of Primary Flow	3.5.10

Table 3.2-4: Initiating Event Type Not Explicitly Addressed in PSAR

Initiating Event ID	Initiating Event Description	Initiating Event ID	Initiating Event Description
MCRW	Multi-module control rod group withdrawal	MLOIA	Multi-module loss of instrument air
SRT	Single module reactor trip	MPLHS	Multi-module partial loss of condenser heat sink
MDRT	Multi-module double reactor trip	MTLHS	Multi-module total loss of condenser heat sink
MPRT	Multi-module plant-wide reactor trip	SFLDBNI	Single module feedwater line break downstream flow element in NIAS
MFWIS	Single valve feedwater header imbalance	SFLUBNI	Single module feedwater line break upstream flow element in NIAS
MFWID	Double valve feedwater header imbalance	SFLSBRB	Single module feedwater line small break in RB
MFWIT	Triple valve feedwater header imbalance	SFLMBRB	Single module feedwater line medium break in RB with induced medium HPB depressurization only on module #1
MSTIS	Single valve steam header imbalance	SFLLBRB	Single module feedwater line large break in RB with induced large HPB depressurization only on module #1
MSTID	Double valve steam header imbalance	SSLBNI	Single module main steam line break in NIAS
MSTIT	Triple valve steam header imbalance	SSLSBRB	Single module main steam line small break in RB
MPLFW	Multi-module partial loss of steam generator feedwater flow	SSLMBRB	Single module main steam line medium break in RB
SPLPF	Single module partial loss of primary flow	SSLLBRB	Single module main steam line large break in RB
MPLPF	Multi-module partial loss of primary flow		

3.2.1.2 Summary Evaluation of DBAs

As described in RG 1.253, NEI 21-07, section C.3 provides an acceptable method for developing the summary of LBEs (AOOs, DBEs, BDBEs, and DBAs) in the safety analysis report. NEI 21-07, section 3.2.2, states:

> … a summary of the evaluation of DBAs presented. This summary should include:
>
> - A table that shows the mapping of DBEs into DBAs with brief word descriptions of the DBEs and DBAs.
>
> - A table that shows the dose consequences of the DBAs for comparison against the 25 rem (0.25 Sv) criterion derived from 10 CFR 50.34 (some DBAs may have no releases and, therefore, no doses.)

The NRC staff's evaluation of PSAR section 3.2 and LME's implementation of the guidance in NEI 21-07, section 3.2.2, and RG 1.253 is provided in subsections 3.2.1.2.1 and 3.2.1.2.2 of this SE.

3.2.1.2.1 A table that shows the mapping of DBEs into DBAs with brief word descriptions of the DBEs and DBAs

The NRC staff reviewed PSAR table 3.2-4, "Mapping of Non-DBA LBEs to DBAs," which describes the mapping of DBEs, AOOs with 5th percentile frequencies in the DBE range, and BDBEs with 95th percentile frequencies in the DBE range to DBAs. Additionally, PSAR section 3.2.2.3, "Summary Evaluation of DBAs," clarifies that PSAR table 3.2-4 includes a seismic DBA which is not mapped from a DBE because the CP stage PRA does not include external hazards, but is evaluated because it is anticipated to be one of the limiting DBAs in terms of radiological consequences. The NRC staff reviewed PSAR table 3.2-4 to ensure that all relevant LBEs are appropriately mapped to a DBA. The NRC staff compared PSAR table 3.2-4 to PSAR table 3.2.2 and associated figures. Based on the NRC staff's review, the NRC staff determined:

- PSAR table 3.2-4 maps twenty-three AOOs to a DBA because the 5th percentile frequency for the AOO crosses into the DBE region. Table 3.2-5 of this SE shows the mapping of these AOOs.

- PSAR table 3.2-4 maps thirty BDBEs to a DBA because the 95th percentile frequency for the BDBE crosses into the DBE region. Table 3.2-5 of this SE shows the mapping of these BDBEs.

- PSAR table 3.2-4 maps the DBEs as follows and table 3.2-5 of this SE shows the mapping of these DBEs.

The NRC staff finds the mapping of the AOOs, DBEs, and BDBEs to a DBA, as shown in table 3.2-5 of this SE, to be reasonable because these LBEs include the same RSF challenges of retaining radionuclides in fuel particles and pebbles, controlling reactivity, and controlling heat removal as that of the selected DBA. The NRC staff finds that the applicant adequately mapped

DBEs to DBAs because the applicant is consistent with NEI 18-04 by: (1) considering BDBEs at the 95th percentile that should be evaluated as a DBE and mapped to a DBA, (2) considering AOOs at the 5th percentile frequency that should be evaluated as a DBE and mapped to a DBA, and (3) mapping all DBEs to DBAs. Based on the information provided in PSAR tables 3.2-4 and 3.2-5, the NRC staff finds that the applicant's brief word descriptions of the AOOs, DBEs, BDBEs, and DBAs are reasonable because the information provided is consistent with NEI 21-07 Section 3.2.2, as endorsed by RG 1.253, and describes the AOOs, DBEs, and BDBEs.

3.2.1.2.2 A table shows the dose consequences of the DBAs for comparison against the 25 rem (0.25 Sv) criterion derived from 10 CFR 50.34

PSAR table 3.2-5 provides the following 30 day mean dose consequences and the 95th percentile dose consequences at the EAB for the DBAs:

DBA	DBA 30-day EAB dose (rem [Sv])	DBA 95th 30-day EAB dose (rem [Sv])	Number of Modules experiencing the same type of event
SCRW-DBA	0.184 (0.00184)	0.214 (0.00214)	4
MTLPF-DBA	0.187 (0.00187)	0.217 (0.00217)	4
SD-DBA	2.410 (0.2410)	2.693 (0.02693)	1
MD-DBA	3.680 (0.3680)	5.907 (0.05907)	1
MTFLW-DBA	0.159 (0.00159)	0.188 (0.00188)	4
MFLB-DBA	0.159 (0.00159)	0.189 (0.00189)	4
MSLB-DBA	0.159 (0.00159)	0.189 (0.00189)	4
SGTL-DBA	0.227 (0.00227)	0.252 (0.00252)	1
SGTR-DBA	0.227 (0.00227)	0.252 (0.00252)	1
Seismic B-DBA	14.4 (0.14400)	23.2 (0.23200)	4

The NRC staff finds that PSAR table 3.2-5 is consistent with NEI 21-07, Section 3.2.2 because the table shows dose consequences of DBAs for comparison against the 25 rem (0.25 Sv) criteria derived from 10 CFR 50.34.

Table 3.2-5: DBAs Presented in PSAR and Associated Mapped LBEs

DBA	Initiator	PSAR Section	AOOs mapped to DBA	DBEs mapped to DBA	BDBEs mapped to DBA
SCRW DBA	Control Rod Withdrawal	3.6.1	SCRW-01	MCRW-01, MCRW-07, SCRW-07, SCRW-13	SCRW-02
MTLPF DBA	Loss of Primary Flow	3.6.2	MDRT-01, MLOOP-01, MPLPF-01, MPRBC-01, MPRBC-02, MPRT-01, MPRT-07, SPLPF-01, SPLPF-02, SPLPF-08	MDRT-07, MDRT-18, MLOOP-03, MLOOP-13, MLOOP-22, MLOOP-23, MPLPF-08, MPLPF-14, MPRT-08 MTLPF-07, MTLPF-22, MTRBC-01, SPLPF-03, SPLPF-04, SPLPF-14, SPLPF-19, SRT-002, SRT-003 SRT-004, SRT-014, SRT-022, STLPF-22	MDRT-08, MDRT-27, MPLPF-03, MPRBC-09, MPRT-09, SPLPF-23, SPLPF-24, SPLPF-28
SD DBA	Small Helium Pressure Boundary Depressurization	3.6.3	SD-068	SD-076, SD-084, SD-098	SD-074, SD-106
MD DBA	Medium Helium Pressure Boundary Depressurization	3.6.4	-	MD-63, MD-68, MD-73, SFLMBRB-01	-

DBA	Initiator	PSAR Section	AOOs mapped to DBA	DBEs mapped to DBA	BDBEs mapped to DBA
MTLFW DBA	Loss of Feedwater	3.6.5	MPLFW-01, MPLFW-02, MPLHS-01, MTLFW-01, MTLHS-06, MTT-01, MTT-02, MTT-08	MFWIS-06, MFWIS-06, MFWIS-11, MLOIA-02, MLOIA-03, MLOIA-07, MLOIA-11, MPLHS-08, MSTIS-01, MSTIS-07, MTLFW-02, MTLHS-02, MTLHS-07, MTT-09	MLOIA-08, MPLFW-09, MTFLW-03, MTLHS-03, MTLHS-11, MTT-13
MFLB DBA	Feedwater Line Break	3.6.6	-	MFLB-01, SFLUBNI-01	MFLB-02, SFLSBRB-01, SFLUBNI-07
MSLB DBA	Main Steam Line Break	3.6.7	MSLB-01	SSLBNI-01, SSLBNI-16, SSLBNI-22, SSLSBRB-01	MSLB-08, SSLSBRB-15
SSGTL DBA	Steam Generator Tube Leak	3.6.8	SSGTL-01, SSGTL-03	SSGTL-07, SSGTL-13, SSGTL-14, SSGTL-18	SSGTL-02, SSGTL-21
SSGTR DBA	Steam Generator Tube Rupture	3.6.9	-	SSGTR-01, SSGTR-03, SSGTR-07, SSGTR-13, SSGTR-14	-
Seismic DBA	Seismic	3.6.12	-	MDRT-23, SFLDBNI-01, SFLDBNI-16, SRT-018	MPLPF-19, MTLHS-21, MTT-24, SCRW-18, SFLDBNI-22, SFLLBRB-01

3.2.1.3 Control Room Dose

PSAR section 3.2.2.2 states: “Consistent with PDC 19, control room doses for all AOOs and DBEs are less than 5.0 rem (0.05 Sv) TEDE over the 30-day period.” The NRC staff’s evaluation of PSAR section 3.2.2.2 is in section 3.1.5 of this SE.

3.2.1.4 Conformance with Offsite Dose Limits

PSAR section 3.2.3, "Conformance with Offsite Dose Limits," states that:

- To address the requirements of 10 CFR 50.34(a)(1)(ii)(D), the applicant applies Option 1 from RG 1.253, section C.3.b, which describes the use of DBA dose consequence results to establish the acceptability of the EAB and LPZ.
- For the LMGS plant, the EAB and LPZ are the same boundary, as defined in PSAR section 2.1.3.3.
- No BDBE or DBA results in a dose greater than 25 rem (0.25 Sv) (30-day TEDE).

The NRC staff's evaluation of the mechanistic source terms and radiological consequences are in sections 3.1.4 and 3.1.5 of this SE, respectively. With all non-DBA LBE 95th percentile consequences within the F-C target, all DBA dose consequences below 10 CFR 50.34 dose limits, and all cumulative risk metrics within the criteria outlined in NEI 18-04, the NRC staff determined that the preliminary design for the LMGS conforms with offsite dose limits.

Additionally, in part V of the LMGS CP application, "Non-Applicabilities and Exemptions," section 2.2.2, "10 CFR 50.34(a)(1)(ii)(D) – Analysis of Postulated Fission Product Release," LME requests an exemption from the requirement to analyze an assumed fission product release associated with an accident that results in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products. This exemption is evaluated in section 15.4 of this SE.

RG 1.253, section C.3.b states: "The uncertainty analyses for the mechanistic source terms and radiological doses should be described as part of the evaluation of conservative assumptions used in the DBA analysis." The NRC staff's evaluation of the mechanistic source terms and radiological consequences are in sections 3.1.4 and 3.1.5 of this SE, respectively.

3.2.2 Conclusion

The NRC staff reviewed the LBE summary information described in PSAR section 3.2 and determined that the identification and classification of LBEs is consistent with NEI 18-04 and RG 1.233. The NRC staff finds that the LBE information provided in the PSAR: (1) is consistent with the guidance in RG 1.253 and NEI 21-07 and (2) provides a complete, consistent, and integrated summary of the ESs that form a large portion of the plant's design bases. Therefore, the NRC staff finds that the LBE summary information meets the requirements of 10 CFR 50.34(a)(1) and (4) for the issuance of a CP in accordance with 10 CFR 50.35.

3.3 Anticipated Operational Occurrences

The NRC staff reviewed PSAR section 3.3, "Anticipated Operational Occurrences," which presents the evaluation of representative or bounding AOOs for the LMGS. Table 3.2-3 of this SE lists the AOOs presented in PSAR section 3.3 with their PRA ES identification and event initiator.

PSAR section 3.2 discusses LBEs which are not explicitly evaluated in PSAR section 3.3 as AOOs. While some of the LBEs discussed in PSAR section 3.2 are represented by LBEs with

similar ES, other LBEs, as discussed in PSAR table 3.2-7, do not have an explicit evaluation performed in the PSAR, which includes the AOOs. In PSAR section 3.2, the applicant committed to performing a fully refined LBE evaluation and a final ES family grouping in support of a future OL application. The NRC staff notes that the PRA methodology for the AOO selection is limited to preliminary design for full-power operations at the CP application stage. As the PRA is expanded beyond full power operation core events, the events analyzed and their outcomes may lead to a different set of LBEs to be identified as AOOs at the OL application stage.

The NRC staff's review and acceptability of the identification of representative or bounding AOOs is presented in section 3.2 of this SE.

The applicable regulatory requirements for the evaluation of the AOOs are as follows:

- 10 CFR 50.34(a)(1) and (4); and
- 10 CFR 50.35(a)(3).

The applicable guidance for the evaluation of the LBE summary are as follows:

- RG 1.233, Revision 0, which endorses NEI 18-04, Revision 1, with clarifications; and
- RG 1.253, Revision 0, which endorses NEI 21-07, Revision 1, with clarifications and additions.

3.3.1 Technical Evaluation

The NRC staff evaluated the information in PSAR section 3.3 against the following criteria for LBE evaluations for a CP application provided in RG 1.253, section C.4:

1. The design will provide sufficient safety margins during normal operations and transient conditions (evaluated in section 3.3.1.1 of this SE).
2. The applicant has identified the SSCs necessary to prevent accidents and mitigate accident consequences (evaluated in section 3.3.1.2 of this SE).
3. The applicant has demonstrated an understanding of the uncertainty associated with the performance of SSCs necessary to prevent accidents and mitigate accident consequences (evaluated in section 3.3.1.3 of this SE).

3.3.1.1 Evaluation of safety margins during normal operations and transient conditions

The NRC staff evaluated the assessment of safety margins during AOOs by reviewing the EM, input parameters and initial conditions, and analysis results.

<u>Evaluation Model</u>

PSAR section 3.1.3 states that the thermodynamic behavior of the plant during LBEs is evaluated using Flownex and GOTHIC and also references the GFQ TR. As described in sections 3.1.2 and 3.1.3 of this SE, the NRC staff finds the use of the TSAM TR and the GFQ

TR acceptable for the preliminary safety analysis of the design and performance of LMGS SSCs in accordance with 10 CFR 50.34(a)(4). An evaluation of the L&Cs associated with TRs along with the acceptability of method of addressing the L&Cs is presented in section 3.1.3 of this SE.

Input Parameters and Initial Conditions (General)

PSAR table 3.3-1 provides the initial plant conditions for LBE calculations. The NRC staff evaluated the initial plant conditions for LBE calculations by comparing the initial plant conditions against design information provided in the PSAR (either directly or IBR). Based on this comparison:

- PSAR table 3.3-1 provides the core thermal power used in LBE evaluations as 199.6 MWt which reflects an uncertainty of 0.3 MWt or 0.15 percent uncertainty. Note 5 of PSAR table 3.3-1 provides clarification that the core thermal power uncertainty does not reflect power measurement uncertainty, rather it reflects small power oscillations that are observed in the steady-state calculations. The NRC staff finds core thermal power used in LBE evaluations to be acceptable because it is consistent with the design information presented in the CDMA TR, which was found acceptable in section 3.1.6 of this SE.

- PSAR table 3.3-1 provides the reactor inlet temperature used in LBE evaluations as 260.0 +/- 0.1 ºC. Note 5 of PSAR table 3.3 provides clarification that the inlet temperature uncertainty does not reflect temperature measurement uncertainty, rather it reflects small temperature oscillations that are observed in the steady-state calculations. The NRC staff finds the reactor inlet temperature used in LBE evaluations to be acceptable because it is consistent with the information presented in the CDMA TR, which was found acceptable in section 3.1.6 of this SE.

- PSAR table 3.3-1 provides the reactivity worth of rod withdrawal as 1519-1522 pcm without including a basis for this value. During the regulatory audit, LME provided documentation to support the calculation of the reactivity worth of rod withdrawal value (Document number 006529, “Preliminary Scoping VSOP Outputs for Safety Analysis”). The NRC staff examined the methodology outlined in this document to calculate the reactivity worth of rod withdrawal value. The NRC staff confirmed that, based on the information provided during the regulatory audit, the withdrawal of 9 RCS rods results in approximately 1520 pcm of reactivity worth. The NRC staff therefore determined that the reactivity worth of rod withdrawal of 1519 - 1522 is an acceptable initial plant condition because it is consistent with the information presented in the CDMA TR, which was found acceptable in section 3.1.6 of this SE.

- PSAR table 3.3-1 provides values for average fuel temperature and average moderator temperature and notes that these conditions in the reference reactor model reflect graphite conditions at 30 years and 60 years of neutron fluence for the outer, upper and lower reflectors. The NRC staff finds the input used for average fuel temperature and average moderator temperature acceptable for preliminary analysis of the LMGS because the temperature values are consistent with the information presented in the CDMA TR, and the expected for graphite lifetimes for the Xe-100 reactor design are consistent with the information in Graphite Core Assembly Material Qualification and Design Methodologies TR, which is also IBR and found acceptable in SE section 6.4.4.

- PSAR table 3.3-1 provides the total worth of RSS and RCS rod insertion as 10305 - 10308 pcm. The NRC staff compared these values to the information provided in table 4, "Typical Control and Shutdown Reactivity Worth," from TR 006889-A, "Reactor Core Design Methods and Analysis," (ML25262A123), which is IBR into the PSAR, and observed that the values provided in PSAR table 3.3-1 are smaller in magnitude (i.e., more conservative). Additionally, the NRC staff examined the calculations supporting PSAR table 3.3-1 and noted that the values provided in PSAR table 3.3-1 correspond to an earlier version of a design calculation than that used for the TR. Based on the information provided in the PSAR, the NRC staff finds the input used for the total worth of RSS and RCS rod insertion acceptable for the preliminary analysis of the LMGS because these values are representative of the values expected for the Xe-100 reactor design presented in the CDMA TR.

- PSAR table 3.3-1 states that material properties are obtained using 30 years of fluence for inner reflector and 60 years for outer, top and bottom reflectors. During the regulatory audit, the NRC staff was able to confirm that aging effects, particularly thermal conductivity, have been modeled using **[[**
  ]] Thermal conductivity is further reduced by 20 percent in DBA calculations to account for uncertainties in the graphite conductivity correlation, fluence, and irradiation temperature. The NRC staff finds the reduction in material thermal conductivity acceptable since the reduced thermal conductivity for the DBA calculations will lead to conservative temperatures and dose consequences.

- PSAR table 3.3-1 does not provide kinetics parameters, radial and axial power distribution, and temperature coefficients of reactivity. However, PSAR section 3.1.6 clarifies that the CDMA TR is used for, "Inputs to the thermodynamics models [used for LBE evaluations …] including kinetics parameters, radial and axial power distribution, temperature coefficients of reactivity, and control and shutdown rod worths." Additionally, the CDMA, as described in section 3.1.6 of this SE and IBR in PSAR table 1.4-4, includes representative reactor physics parameters for the Xe-100 reactor design. Specifically, section 4, "Results from Core Physics Models," of the CDMA TR includes kinetics parameters, power distributions, and temperature coefficients of reactivity. The NRC staff finds that the non-DBA LBEs are evaluated using reasonable values for kinetics parameters, power distributions, and temperature coefficients of reactivity because these values were obtained using a methodology approved for the preliminary design along with a plan for V&V to be performed prior to the OL application stage, which is found acceptable by the NRC staff in SE section 3.1.6.

- PSAR table 3.3-1 states that decay heat is input as a curve fit to DIN 25485 standard results. The use of DIN 25485 for modeling decay heat in the Xe-100 reactor design is addressed in the NRC staff's SE on CDMA (ML25262A123) which states:

  > While the use of DIN 25485 standard for calculating decay heat is reasonable based on its application in other reactors and core physics codes, as stated by the licensee, the NRC staff notes that additional details are needed for how the calculation is performed for a validation basis for the standard DIN 25485 for use in Xe-100 applications. Consistent with the requirements of 10 CFR 50.34(a)(4), the NRC staff does not expect the details on calculation of decay heat using DIN 25485 and its validation for use in Xe-100 applications

for the preliminary design but expects the final design methodology to include such details.

Based on the information provided in PSAR table 3.3-1 and the use of CDMA, the NRC staff finds that the decay heat modeling is acceptable for the preliminary analysis of LBEs because the modeling is being applied consistently with an approved methodology.

- The NRC staff finds that the remaining inputs listed in PSAR table 3.3-1, helium mass flow (total and pebble bed), reactor pressure, and main steam pressure, are reasonable because they are consistent with the values for the LMGS design presented in the CDMA TR. The pebble bed mass flow represents approximately 10.5 percent bypass flow which the applicant attributed to their engineering judgement. The NRC staff finds pebble bed bypass flow to be a reasonable value for the preliminary analysis based on the literature review of current and past pebble bed high-temperature gas-cooled reactor (HGTR) designs.

PSAR section 3.3 states that: "[PSAR] Table 3.3-2 provides the initial circulating and deposited isotopic inventory within the HPB for all LBEs unless otherwise noted in an LBE description." The description states that the inventory was calculated using XSTERM, which incorporates the Point Kinetics Core Simulation Model and HPBM for steady-state hot full power core and for fuel performance calculations respectively, as described in TR 000632, "Xe-100 Licensing Topical Report Mechanistic Source Term Approach," Revision 3, which is IBR.

The NRC staff audited the calculation that provides the radionuclide inventory provided in PSAR table 3.3-2 to confirm the results for source term generation. Further, the NRC staff performed confirmatory calculations using SCALE to verify these results. The acceptability of the radionuclide inventory for the Xe-100 design is discussed in section 3.1.4 of this SE.

The NRC staff notes that the uncertainty bands on the initial plant conditions for the LBEs in PSAR table 3.3-1 are based on best estimate plus uncertainty, and do not include the operating band for the initial plant conditions. The operating band is selected to provide operational flexibility. The plant safety basis calculations are performed for the operating band to ensure that the plant can be operated safely within the defined limits. Further, some of the uncertainties on initial plant conditions described in PSAR table 3.3-1 do not include computational uncertainties (e.g. rod worth calculations).

While the evaluation of LBEs using initial conditions presented in PSAR table 3.3-1 provides reasonable assurance of safety for the preliminary design at the CP stage, an evaluation of operating band for the initial plant conditions as well as comprehensive analysis on the treatment of uncertainties is needed to ensure LBE evaluations yield conservative results for 95^{th} percentile consequence. The applicant provided a commitment to perform uncertainty quantification at the OL application stage.

Based on the information provided in PSAR table 3.3-1 and evaluated above, the NRC staff finds that the generic LBE inputs are acceptable for the preliminary LMGS design.

AOO Specific Evaluation and Results

The NRC staff's evaluation of modeling of AOO specific plant conditions and results are presented below. For each AOO, the NRC staff ensured that, consistent with RG 1.233: (1) the event initiator and the ES narrative in the PSAR is sufficient to understand the plant response and identify the SSCs credited to mitigate the event, and (2) the dose consequences and the details of the radionuclide release are provided, along with the key parameters of interest, and demonstrate margin to the F-C curve.

The NRC staff notes that the ES description and the end state in PSAR section 3.3 may be different than the actual plant response provided for the AOO in PSAR table 3.2-2. This is attributed to LME describing the consequence analysis sequence in PSAR section 3.3, rather than in the ES. In many cases, LME applied conservative assumptions that would represent additional failures which are reflected by the differences in the ES descriptions between PSAR table 3.2-2 and section 3.3. The NRC staff finds this acceptable since the ES descriptions in PSAR table 3.2-2 were confirmed to be accurate and LME confirmed in the audit that the differences were due to the conservatism used in the event modeling.

The NRC staff's review of each AOO evaluation is documented below.

Turbine Trip AOO

PSAR section 3.3.1 describes the TT AOO analysis and the events composing PRA sequences MTT-01 and MTT-02. PSAR tables 3.3.1-1 and 3.3.1-2 show the sequence of events for MTT-01 and MTT-02 events, respectively.

The IE for this AOO is trip of both turbine generators from an initial steady-state full-power condition. For the MTT-01 event, the distributed control system (DCS) decreases the reactor power to a new equilibrium at 40 percent of rated steam flow. The MTT-02 event models a failure of DCS to maneuver the plant to new lower steady-state with the investment protection system (IPS) providing actuation signal to initiate controlled shutdown. The MTT-02 event ends in both control rod and shutdown rods being inserted to control reactivity, and cooling maintained by the main feedwater pumps via SG header and condenser. PSAR tables 3.3.1-1 and 3.3.1-2 detail the sequence of events for MTT-01 and MTT-02 events, respectively.

The results presented for the events show no release of radionuclides leading to no dose consequence. PSAR figures 3.3.1-1 through 3.3.1-6 show the plots for reactor power, mass flow rate, rod position and associated reactivity for both MTT-01 and MTT-02.

Main Steamline Break AOO

PSAR section 3.3.2 describes the MSLB AOO analysis and the events composing PRA sequence MSLB-01.

The IE for this AOO is a guillotine break downstream of the tee joining the supply lines from two separate SG tubes. A break at this location maximizes the rupture size and affects both tube bundles. Following event initiation, SG temperature and pressure drop resulting in an increase in helium circulator speed in response to reduced steam pressure, and a closing of feedwater regulating valves in response to reduced steam temperature. An RT, initiated by RPS or IPS on high helium to feedwater flow ratio, results in the insertion of control rods and shutdown rods,

and trips the helium circulators of the helium circulator system (HCS). Following the trip, DCS initiates operation of start-up and shutdown system (SSS). Residual heat removal is established through the SSS by restarting a helium circulator.

The results presented for the MSLB AOO show no release of radionuclides leading to no dose consequence. Consistent with the guidance provided in NEI 21-07, section C-3.3, since the event does not lead to release of radionuclides, the applicant did not identify key reactor parameters associated with the MSLB AOO event progression.

Control Rod Bank Withdrawal AOO

PSAR section 3.3.3 describes the CRW AOO analysis and the events composing PRA sequence SCRW-01.

The IE for this AOO is withdrawal of a full control rod bank, consisting of nine rods, from a module at full power at a rate of 1.2 m/sec. The possible causes for the IE are listed as a spurious rod withdrawal signal from IPS or DCS, or human error. The PSAR states that the withdrawal rate is 20 percent faster than system design. Following event initiation, an RT is initiated by RPS on high neutron flux resulting in the insertion of control rods and shutdown rods. The helium circulators do not trip for this event, and main loop cooling is maintained through continued operation of the circulators. The event ends with the reactor shutdown to decay power level with intact HPB and helium cooling provided to the core.

The results presented for the SCRW AOO event show no release of radionuclides leading to no dose consequence. Consistent with guidance provided in NEI 21-07, section C-3.3, since the event does not lead to release of radionuclides, the applicant did not identify key reactor parameters associated with the SCRW AOO event progression.

Small HPB Depressurization AOO

PSAR section 3.3.5 describes the SD AOO analysis and the events composing PRA sequence SD-68.

The IE for this AOO is a 5 mm break at the reactor inlet cross-vessel duct. The PSAR states that a 10 mm break at the same location was also analyzed (PRA sequence SD-68-1) but the 5 mm break proved to be limiting due to the smaller break size resulting in higher fuel temperatures over a longer period. Following event initiation, the HPB begins to depressurize leading to RPS actuation on low pressure which results in the insertion of control rods and shutdown rods approximately nine minutes after event initiation. The helium circulators continue to maintain decay heat removal via circulator runback as controlled by DCS. Once the surge margin drops below 5 percent, a single circulator is tripped, and the other circulator keeps operating within design margins. Helium flow through the leak continues for approximately 20 hours and is arrested when the pressure drops to atmospheric pressure. The reactor shuts down and remains shut down for rest of the event with fuel temperatures well below the DLOFC event described in PSAR section 3.4.8 for long-term cooling. PSAR figures 3.3.5-1 through 3.3.5-5 demonstrate the above ES.

The NRC staff notes that while the plant response for the event in PSAR table 3.2-2 shows the transient ending in forced cooling maintained by feedwater pumps, the consequence analysis sequence in PSAR section 3.3 models long-term DLOFC cooling with passive RCCS for the

event. The NRC staff finds this to be acceptable since this conservative assumption represents additional failures and leads to bounding dose consequences.

The NRC staff audited the supporting calculation and noted that SD events are run up to a break size of 10 mm in diameter, which corresponds to the size of the small instrument line in the HPB and confirmed that 5 mm diameter break remains bounding.

The SD AOO has a total 30-day radiation dose (TEDE), including the dose during the long-term cooldown, less than 2.5 mrem (0.025 mSv) for the mean dose. However, the 95th percentile dose is greater than 2.5 mrem (0.025 mSv) as shown in PSAR table 3.2-2. The control room dose is less than 5 rem (0.05 Sv). PSAR figures 3.3.5-1 through 3.3.5-5 show the plots for reactor power, flow rate, HPB pressure and associated mass flow rate for the 5 mm break (SD-68-0) event. Plots of key parameters for this AOO are also provided in PSAR section 3.5.2 for comparison purposes.

The NRC staff notes that the tables of radionuclides to describe the MST is not provided for this event, which is not consistent with guidance provided in NEI 21-07, section C-3.3; however, the NRC staff finds it acceptable that the MST is not provided for this individual ES for the CP stage since there is a table of radionuclides provided for an event with a similar event progression (i.e., SD-74 BDBE sequence) in PSAR section 3.5.2. The primary difference between the SD-74 BDBE and the SD-68 AOO is that the forced circulation stops earlier in the BDBE sequence, resulting in higher fuel temperatures during the depressurization, and therefore higher offsite dose. Thus, the NRC staff has reasonable assurance that the event consequences for SD-68 are sufficient to assure safety margins during normal operations and transient conditions for the preliminary LMGS design.

Loss of Steam Generator Feedwater AOO

PSAR section 3.3.6 describes the loss of SG feedwater (LOFW) AOO analysis and the events composing PRA sequence MTLFW-01.

The IE for this AOO is tripping of the high-pressure feedwater pump (HPFWP) motors, which in turn reduces the SG feedwater flow and results in stalling of primary to secondary side heat transfer. An RT, initiated by RPS and IPS on high helium to feedwater flow ratio, results in the insertion of control rods and shutdown rods, and trips the helium circulators. Following the trip, DCS initiates the operation of SSS. Residual heat removal is established through SSS by restarting a helium circulator. The event ends with the reactor shutdown to decay power level with intact HPB and partial helium cooling provided to the core.

The MTLFW AOO has a total 30-day radiation dose (TEDE), including the long-term cooldown, less than 2.5 mrem (0.025 mSv) for the mean dose, however the 95th percentile dose is greater than 2.5 mrem (0.025 mSv) as shown in PSAR table 3.2-2. The control room dose is less than 5 rem (0.05 Sv).

The applicant did not provide plots for key reactor parameters for this AOO. While this is not consistent with guidance provided in NEI 21-07, section C-3.3, the NRC staff finds it acceptable that the applicant has not provided plots for this individual AOO for the CP stage since the PSAR section 3.5.5 describes MTLFW-03 BDBE, which has a similar sequence with additional failures, resulting in a higher dose consequence, and is further mapped into a DBA in PSAR section 3.6.5. Thus, the NRC staff has reasonable assurance that the event consequences for

MTLFW-01 are sufficient to assure safety margins during normal operations and transient conditions for the preliminary LMGS design.

Steam Generator Tube Leak AOO

PSAR section 3.3.7 describes the steam generator tube leak (SGTL) AOO analysis and the events composing PRA sequence SSGTL-01.

The IE for this AOO is a 3.16 mm break in a SG tube, which represents the geometric mean of the defined SSGTL range of 1 mm to 10 mm diameter. PSAR section 3.6.8 states that the SSGTL DBA analysis uses the upper bound value of 10 mm instead of the geometric mean.

An RT initiated by RPS and IPS on moisture, results in the insertion of control rods and shutdown rods. Following this, SG isolation is initiated by RPS and IPS. The SG isolation proceeds first with feedwater-side isolation followed by the steam-side isolation and then the SSS blowdown header valves open. This leads to SG inventory blowing down into the SSS tank until the target pressure is reached, then the valves reclose. After SG isolation completes, the leak in the affected SG bundle continues until flow into the HPB and is arrested by pressure equalizing between the affected bundle and the HPB. The operator takes actions to initiate SSS cooling once the SSS blowdown valves are closed and there is a pressure difference of at least 0.1 MPa (145 psi) between the A and B steam lines. The difference in pressure between the steam lines provides a mechanism for detecting the leak in the affected bundle.

The event ends with the reactor shutdown to decay power level with intact HPB and partial helium cooling provided to the core. The event ends with the reactor shut down and decay heat removal via the SSS. There are no relief valve actuations during the event. The event increases the circulating inventory due to steam wash-off effects. PSAR table 3.3.7-3 shows that the total cumulative graphite oxidation is limited to 1.9 g of graphite. The NRC staff notes that the applicant identifies an operator action for restarting the SSS cooling in the intact bundle for this event.

The SSGTL AOO has a total 30-day radiation dose (TEDE), including the long-term cooldown, less than 2.5 mrem (0.025 mSv) for the mean dose, however the 95th percentile dose is greater than 2.5 mrem (0.025 mSv) as shown in PSAR table 3.2-2. The control room dose is less than 5 rem (0.05 Sv). PSAR table 3.3.7-1 provides initial plant conditions for all SSGTL and SSGTR LBEs. PSAR table 3.3.7-2 provides the operational status of SSCs for all SSGTL and SSGTR LBEs with steam and oxidation levels provided in PSAR table 3.3.7-3. PSAR table 3.3.7-4 provides the transition points for event analysis to long-term cooling in Flownex for all LBEs and PSAR table 3.3.7-5 provides relief valve actual status.

The applicant did not provide plots for key reactor parameters for the event. While this is not consistent with guidance provided in NEI 21-07, section C-3.3, the NRC staff finds it acceptable at the CP stage for the applicant to use the results from a different event, SSGTL-21 BDBE, as representative for this event. PSAR section 3.5.8 describes SSGTL-21 BDBE, which has a similar sequence with additional failures, leading to higher dose consequence, and is further mapped into a SSGTL DBA in PSAR section 3.6.8. Thus, the NRC staff has reasonable assurance that the event consequences for SSGTL-01 are sufficient to assure safety margins during normal operations and transient conditions for the preliminary LMGS design.

Loss-of-Offsite Power AOO

PSAR section 3.3.8 describes the LOOP AOO and the events composing PRA sequence MLOOP-01.

The IE for this AOO is LOOP upon which three reactor modules trip via RPS and IPS, while one remains available to supply LMGS house loads until the cause of power loss is remedied. The event ends with three reactor modules shutdown to decay power level with intact HPB and helium cooling provided to the core. If none of the turbines remain online, the applicant assumes that the plant standby diesel generators will be used to provide forced cooling via SSS for all four modules.

The results presented for the MLOOP AOO event show no release of radionuclides leading to no dose consequence. Consistent with guidance provided in NEI 21-07, section C-3.3, since the event does not lead to release of radionuclides, the applicant did not identify key reactor parameters associated with the MSLB event progression.

Summary AOO Evaluations and Safety Margins

The NRC staff finds that the evaluation of AOOs in PSAR section 3.3 provides reasonable assurance of safety margins during AOOs because:

1. The applicant used an acceptable EM for the evaluations.

2. The applicant used suitably conservative inputs and boundary conditions for the evaluations.

3. The applicant describes the results of the evaluations including the event initiator, and the narrative of each ES is sufficient to understand the plant response and identify the SSCs credited to mitigate the event.

4. The applicant provides the dose consequences and the details of the radionuclide release where applicable, along with the key parameters of interest to demonstrate margin to the F-C curve.

For the events where the dose consequence, details of the radionuclide release or the key parameters of interest are not provided at the CP stage, the NRC staff identified events in other sections of the PSAR that have a similar event progression, with additional failures, that would lead to representative or bounding and dose consequences.

3.3.1.2 Identification of SSCs necessary to prevent and mitigate AOOs

Based on the event descriptions provided in PSAR section 3.3, the NRC staff identified the SSCs credited to prevent and mitigate AOOs. Table 3.3-1 of this SE documents the results of the NRC staff's review of AOOs.

Table 3.3-1: SSCs Credited to Prevent Accidents and Mitigate AOOs

PSAR Section	AOO	Initiator	SSCs Credited to Prevent and Mitigate
3.3.1	MTT-01, MTT-02	Turbine Trip	MTT-01 Nuclear island electrical system (NIE), nuclear instrumentation system (NIS) DCS, reactivity control and shutdown system (RCSS), HCS, steam generator system (SGS) MTT-02 NIE, NIS, IPS, RCSS, DCS, HCS, SGS
3.3.2	MSLB-01	Main Steamline Break	NIE, NIS, IPS, RPS, RCSS, DCS, SSS, HCS
3.3.3	SCRW-01	Control Rod Withdrawal	NIE, NIS, RPS, RCSS, DCS, HCS
3.3.5	SD-68	Small HPB Depressurization	NIE, NIS, RPS, RCSS, DCS, HCS, RCCS
3.3.6	MTLFW-01	Loss of Feedwater	NIE, NIS, IPS, RPS, RCSS, DCS, SSS, HCS
3.3.7	SSGTL-01	Steam Generator Tube Leak	NIE, NIS, IPS, RPS, RCSS, DCS, SSS, HCS
3.3.8	MLOOP-01	Loss of Offsite Power	NIE, NIS, IPS, RPS, RCSS, SSS, HCS

Based on the information provided in PSAR section 3.3, the NRC staff finds that sufficient information is provided to identify SSCs credited to prevent and mitigate AOOs because the application clearly identified the SSCs credited to prevent and mitigate AOOs using the information provided in the event descriptions.

3.3.1.3 Uncertainty Associated with Performance of SSCs Necessary to Prevent and Mitigate AOOs

RG 1.253, section C.4, “Licensing Basis Events,” describes the connection between evaluating uncertainty associated with performance of SSCs and the identification of safety features and components that require R&D as follows:

> [These criteria] support[s] staff findings associated with 10 CFR 50.35(a)(3), namely, that the application describes the safety features and components that require research and development, and that the applicant will conduct a reasonably designed research and development program to resolve any associated safety questions.

An evaluation of operating band for the input parameters used in LBE evaluation and treatment of uncertainties is needed to ensure LBE evaluations yield conservative results for the 95th percentile consequence is presented in SE section 3.1.1 above. In section 3.1.2.1 of this SE,

the NRC staff concluded that PSAR table 1.4-11 identifies the scope of work needed to address L&Cs associated with the use of the TSAM TR which includes uncertainty quantifications.

The applicant provided a commitment to complete uncertainty quantification at the OL application stage. This provides reasonable assurance that any potential safety questions associated with the adequacy of SSCs and the integral system relied for mitigation of AOOs will be addressed prior to the issuance of an OL.

3.3.2 Conclusion

Based on its review, the NRC staff finds that the description of AOOs provided in PSAR section 3.3 is consistent with the applicable regulatory guidance and meets the regulatory requirements in 10 CFR 50.34(a)(1), 10 CFR 50.34(a)(4) and 10 CFR 50.35(a)(3). This finding is based on the NRC staff's conclusions that the design will provide sufficient safety margins during normal operations and transient conditions, the applicant has identified the SSCs necessary to prevent and mitigate AOOs, and the applicant has demonstrated plan for addressing the uncertainty associated with the performance of SSCs necessary to prevent and mitigate AOO consequences.

3.4 Design-Basis Events

The NRC staff reviewed PSAR section 3.4, "Design Basis Events," which presents the evaluation of representative or bounding DBEs for the LMGS. Table 3.2-3 of this SE lists the DBEs presented in PSAR section 3.4 with their PRA ES identification and event initiator.

Also, PSAR section 3.2 discusses LBEs which are not explicitly evaluated in PSAR section 3.4 as DBEs. While some of the LBEs discussed in PSAR section 3.2 are represented by LBEs with similar ESs, other LBEs, as discussed in PSAR table 3.2-7, do not have an explicit evaluation performed in the PSAR, which includes the DBEs. In PSAR section 3.2, the applicant committed to performing a fully refined LBE evaluation and a final ES family grouping in support of the OL application. The PRA methodology for DBE selection is limited to preliminary design for full-power operations at the CP application stage. As the PRA is expanded beyond full power operation core events, the events analyzed and their outcomes may lead to a different set of LBEs to be identified as DBEs for the OL application.

The NRC staff's review and acceptability of the identification of representative or bounding DBEs is presented in section 3.2 of this SE.

The applicable regulatory requirements for the evaluation of the DBEs are as follows:

- 10 CFR 50.34(a)(1) and (4); and
- 10 CFR 50.35(a)(3).

The applicable guidance for the evaluation of the LBE summary are as follows:

- RG 1.233, Revision 0, which endorses NEI 18-04, Revision 1, with additions; and

- RG 1.253, Revision 0, which endorses NEI 21-07, Revision 1, with clarifications and additions.

3.4.1 Technical Evaluation

The NRC staff evaluated the information in PSAR section 3.4 against the following criteria for LBE evaluations for a CP application provided in RG 1.253, section C.4:

1. The design will provide sufficient safety margins during normal operations and transient conditions (evaluated in section 3.4.1.1 of this SE).

2. The applicant has identified the SSCs necessary to prevent accidents and mitigate accident consequences (evaluated in section 3.4.1.2 of this SE).

3. The applicant has demonstrated an understanding of the uncertainty associated with the performance of SSCs necessary to prevent accidents and mitigate accident consequences (evaluated in section 3.4.1.3 of this SE).

3.4.1.1 Evaluation of safety margins during normal operations and transient conditions

The NRC staff evaluated the assessment of safety margins during DBEs by reviewing the EM, input parameters and initial conditions, and analysis results.

Evaluation Model

The EM for evaluation of the DBEs is identical to the EM described in section 3.3 of this SE for the AOOs. Therefore, the conclusion on the EM in section 3.3 for AOOs is also applicable for DBEs.

Input Parameters and Initial Conditions (General)

The general input parameters and initial conditions in PSAR table 3.3-1 for evaluation of the DBEs are identical to the input parameters and initial conditions described in section 3.3 of this SE for the AOOs. Therefore, the conclusion on the general input parameters and initial conditions in section 3.3 for AOOs is also applicable for DBEs.

DBE Specific Plant Conditions and Results

The NRC staff's evaluation of the DBE specific plant conditions and results are presented below. For each DBE, the NRC staff ensured that, consistent with RG 1.233: (1) the event initiator and the ES narrative in the PSAR is sufficient to understand the plant response and identify the SSCs credited to mitigate the event, and (2) the dose consequences and the details of the radionuclide release are provided, along with the key parameters of interest, and demonstrate margin to the F-C curve.

The ES description and the end state in PSAR section 3.4 may be different than the actual plant response provided for the DBE in PSAR table 3.2-2. This is attributed to LME describing the consequence analysis sequence in PSAR section 3.4, rather than the ES. In many cases, LME applied conservative assumptions that would represent additional failures which are reflected by the differences in the ES descriptions between PSAR table 3.2-2 and section 3.4. The NRC staff

finds this acceptable since the ES descriptions in PSAR table 3.2-2 were confirmed to be accurate and LME confirmed in the audit that the differences were due to the conservatism used in the event modeling.

The NRC staff's review of each DBE evaluation is documented below.

Loss-of-Offsite Power DBE

PSAR section 3.4.1 describes the LOOP DBE analysis and the events composing PRA sequence MLOOP-23.

The initiator for this DBE is a LOOP to all four reactor modules from full-power operating conditions. The PSAR states that this analysis bounds LOOP to a single module. Post event initiation, the turbine trips due to LOOP. An RT, initiated by RPS on high mass flow ratio, results in the insertion of control rods and shutdown rods, and the helium circulator speed decreases to zero following the circulator motor set to a fault condition. Following the trip, DCS initiation of SSS fails. Residual heat removal is established through RCSS in passive mode.

The plot of HPB pressure, as shown in PSAR figure 3.4.1-3, shows that the HPB pressure is gradually increasing at the end of the 24-hour analysis period, but PSAR section 3.4.1 states that the long-term cooling modeled for this event with a revised reflector thermal conductivity model shows that HPB pressure does not exceed the RPS setpoint. Also, the RPV temperature, as shown in PSAR figure 3.4.1-2, decreases in the first few hours of the transient and then begins to rise due to increasing fuel temperatures and lack of helium flow. The event ends with the reactor shutdown to decay power level with intact HPB and long-term cooling (PLOFC) provided by passive RCCS.

The MLOOP DBE has a mean and 95th percentile total 30-day radiation dose (TEDE), including the long-term cooldown, greater than 2.5 mrem (0.025 mSv), as shown in PSAR table 3.2-2. The control room dose is less than 5 rem (0.05 Sv). Details of the radionuclide release for the event, including the long-term PLOFC event, are provided in PSAR table 3.4.1-1. The fuel, moderator, and RPV component temperatures, as well as the HPB pressure for the 24-hour post-RT, are provided in PSAR figures 3.4.1-1, 3.4.1-2, and 3.4.1-3.

The LOOP DBE is mapped to a loss of primary flow (MTLPF) DBA as shown in PSAR table 3.2-4. The acceptability of mapping the LOOP DBE to MTLPF DBA is evaluated in section 3.2 of this SE. PSAR section 3.6.2 describes the MTLPF DBA.

Loss of Primary Flow DBE

PSAR section 3.4.2 describes the LOPF DBE analysis and the events composing PRA sequence MTLPF-22.

The IE for this DBE is the loss of primary flow from full-power operating conditions. The PSAR states that the possible causes for the IE include spurious trips via signals from DCS, IPS or RCS, common cause failures of the circulators or their support systems, or a human operator-induced trip. Following the initiation of event, circulators begin to coast down and DCS reduces the feedwater flow to match the primary flow reduction. An RT, initiated by RPS on high mass flow ratio, results in the insertion of control rods and shutdown rods, and the helium circulators trip. Following an unsuccessful circulator restoration, residual heat removal is established

through active RCSS. The event ends with the reactor shutdown to decay power level with intact HPB and long-term cooling provided by active RCCS. The consequence analysis sequence for this event in PSAR section 3.4 models long-term PLOFC cooling with passive RCCS. The NRC staff find this to be acceptable since this conservative assumption represents additional failures and leads to bounding dose consequences.

The MTLPF DBE has a mean and 95th percentile total 30-day radiation dose (TEDE), including the long-term cooldown, greater than 2.5 mrem (0.025 mSv) as shown in PSAR table 3.2-2. The control room dose is less than 5 rem (0.05 Sv). Details of the radionuclide release for the event, including the long-term PLOFC event, are provided in PSAR table 3.4.2-1. The details of key parameters such as circulator mass flow, feedwater flow, rod position, fuel temperatures, HPB pressure and He/FW flow for the event are provided in PSAR figures 3.4.2-1 through 3.4.2-7.

The MTLPF DBE is mapped to a MTLPF DBA as shown in PSAR table 3.2-4. PSAR section 3.6.2 describes the MTLPF DBA.

Medium HPB Depressurization DBE

PSAR section 3.4.3 describes the MD DBE analysis and the events composing PRA sequence MD-68.

The IE for this DBE is a 65 mm break at the reactor inlet cross-vessel cold-duct. The PSAR states that three break diameters were chosen for the break events at 10 mm, 25 mm, and 65 mm diameters. The 10 mm break corresponds to the small instrument line in the HPB, the 65 mm break corresponds to the largest HPB pipe at the top of refueling tube, and the 25 mm break is the approximate geometric mean of the two other breaks.

Following the initiation of the event, the HPB rapidly depressurizes. An RT, initiated by RPS on low pressure setpoint, results in the insertion of control rods and shutdown rods. Immediately after the rod insertion, the helium circulators trip on a high mass flow ratio signal generated by RPS. The event ends with the reactor shutdown to decay power level with DLOFC long-term cooling provided by passive RCCS.

The plant's response for the event in PSAR table 3.2-2 shows the transient ending in forced cooling maintained by feedwater pumps, but the consequence analysis sequence in PSAR section 3.4 shows long-term DLOFC cooling with passive RCCS for the event. The NRC staff finds this to be acceptable since this conservative assumption represents additional failures and leads to bounding dose consequences.

The MD DBE has a total 30-day radiation dose (TEDE), including the long-term cooldown, less than 2.5 mrem (0.025 mSv) for the mean, however the 95th percentile dose is greater than 2.5 mrem (0.025 mSv) as shown in PSAR table 3.2-2. The control room dose is less than 5 rem (0.05 Sv). Details of the radionuclide release for the event, including the long-term DLOFC event, are provided in PSAR table 3.4.3-1. PSAR section 3.5.3 includes details of key parameters for this DBE and are identified in the figures as 'MD-68-0.' PSAR section 3.5.3 describes the reactor power, fuel temperatures, mass flow rate and core reactivity during the event.

The MD DBE is mapped to corresponding MD DBA as shown in PSAR table 3.2-4. PSAR section 3.6.4 describes the MD DBA.

Main Feedwater Line Break DBE

PSAR section 3.4.4 describes the MFLB DBE analysis and the events composing PRA sequence MFLB-01.

The IE for this DBE is a double guillotine break of the combined feed line header. An RT, initiated by RPS on high cold helium temperature at the SG outlet, results in the insertion of control rods and shutdown rods. Following that, the helium circulators also trip on high cold helium temperature at the SG outlet. Following the trip, DCS initiates SSS. Residual heat removal is established through SSS by restarting a helium circulator.

PSAR figure 3.4.4-1 shows that the helium circulator speed does not go below 500 rpm even after the trip and states that the result shown on the plot is an artifact of a circulator friction model not being included in the Flownex code. The PSAR points to figure 3.4.4-2 where the mass flow rate ceases around 100 seconds. The PSAR also notes that the mass flow increase seen at about 400 seconds represents one circulator restarting to provide cooling through SSS operation.

The MFLB DBE does not result in total radiation dose (TEDE) at the EAB or control room. PSAR figures 3.4.3-1 through 3.4.2-5 provide the details of key parameters such as circulator speed, mass flow rate, HPB pressure, and fuel temperatures for the event.

The MFLB DBE is mapped to corresponding MFLB DBA as shown in PSAR table 3.2-4. PSAR section 3.3.6 describes the MFLB DBA.

Steam Generator Tube Rupture DBE

PSAR section 3.4.5 describes the SGTR DBE analysis and the events composing PRA sequence SSGTR-14. The PSAR states that since the tube rupture event is more limiting than the similar tube leak event, a description of the SSGTL DBE sequences is not provided.

The initiator for this DBE is a 15.8 mm break in a SG tube near the top of SG bundle A. The PSAR states that for events designated as SG tube leaks, a break size of 3.16 mm was used which represents the geometric mean of 1 mm and 10 mm, while the SGTR LBEs used a break size of 15.8 mm which represents the geometric mean of the defined SSGTR range of 10 mm to 25 mm. For the SSGTR DBA, a double-ended guillotine break is modeled as two breaks of 16 mm each.

The PSAR states that the ES and consequences are essentially identical to the SSGTL AOO described in PSAR section 3.3.7, with the primary difference being the assumed failure of the SSS for the DBE. The HPB moisture signals from the IPS and RPS result in the RT leading to the insertion of control and shutdown rods, followed by the circulator trip. The isolation of the affected SG has two steps: feedwater-side isolation and steam-side isolation. Following opening and subsequent closing of the SG dump valves, DCS initiation of the SSS cooling fails. The event ends with the reactor shutdown to decay power level with intact HPB and long-term cooling (PLOFC) provided by active RCCS. The consequence analysis sequence for this event in PSAR section 3.4 models long-term PLOFC cooling with passive RCCS. The NRC staff finds this to be acceptable since this conservative assumption represents additional failures and leads to bounding dose consequences.

The SSGTR DBE has a total mean and 95th percentile 30-day radiation dose (TEDE), including the long-term cooldown, greater than 2.5 mrem (0.025 mSv). The control room dose is less than 5 rem. Details of the radionuclide release for the event, including the long-term PLOFC event, are provided in PSAR table 3.4.5-1. PSAR figures 3.4.5-1 through 3.4.5-6 provide the details of key parameters for this DBE including humidity, circulator mass flow, steam outlet pressure, tube rupture flow, HPB pressure and core reactivity during the event.

The SSGTR DBE is mapped to corresponding SSGTR DBA as shown in PSAR table 3.2-4. PSAR section 3.6.9 describes the SSGTR DBA.

Control Rod Withdrawal DBE

PSAR section 3.4.6 describes the CRW DBE analysis and the events composing PRA sequence SCRW-13.

The PSAR states that the ES and consequences are similar to the SCRW AOO described in PSAR section 3.3.3, with the primary difference being that the main cooling loop is unavailable and the SSS provides cooling. The IE for this DBE is withdrawal of a full control rod bank at full power at a rate of 1.2 m/sec. The PSAR states that the withdrawal rate is 20 percent faster than the system design. Following the event initiation, RT is initiated by RPS on high neutron flux, resulting in the insertion of control rods and shutdown rods. The helium circulators subsequently trip on high feedwater (He/FW) ratio due to the feedwater flow reduction outpacing the coastdown of circulators and DCS automatically initiates cooling on the SSS. The event ends with the reactor shutdown to the decay power level with intact HPB and the SSS providing the cooling.

The PSAR notes that figure 3.4.4-6 shows that the helium circulator speed does not go below 500 rpm even after the trip and states that it is an artifact of a circulator friction model not being included in the Flownex code. The PSAR also notes that mass flow increases once when one circulator restarts to provide cooling through the SSS operation.

The SCRW DBE does not result in total radiation dose (TEDE) at the EAB or control room. PSAR figures 3.4.6-1 through 3.4.6-6 provide the details of key parameters such as rod position, neutron flux, HPB pressure, fuel temperatures, He/FW mass flow ratio and circulator speed for the event.

The SCRW DBE is mapped to corresponding SCRW DBA as shown in PSAR table 3.2-4. PSAR section 3.6.1 describes the SCRW DBA.

Loss of Reactor Building Cooling Water DBE

PSAR section 3.4.9 describes the loss of reactor cooling water (MTRBC) DBE analysis and events composing PRA sequence MTRBC-01.

The IE for this DBE is total loss of RB cooling water where all four equipment trains of RB cooling water fail to operate while at full-power condition. The PSAR states that the possible causes for the IE include common cause failure of pumps or heat exchanges, failure of shared piping or other combinations of independent failures. During the event, the helium circulators and rod drives trip to prevent overheating. The total loss of circulation causes IPS and RPS to initiate RT on flow ratio, leading to insertion of control and shutdown rods. The feedwater pumps

and SSS are both available for the event but are ineffective due to tripping of helium circulators. The event ends with the reactor shutdown to decay power level with intact HPB and long-term cooling (PLOFC) provided by active RCCS. The consequence analysis sequence for this event in PSAR section 3.4 models long-term PLOFC cooling with passive RCCS. The NRC staff finds this to be acceptable since this conservative assumption represents additional failures and leads to bounding dose consequences.

The PSAR states that since the HPB relief setpoint is never breached during the event, there is no release of radionuclides from the transient portion of the DBE. The MTRBC DBE has total mean and 95th percentile 30-day radiation dose (TEDE), including the long-term cooldown, greater than 2.5 mrem (0.025 mSv). The control room dose is less than 5 rem (0.05 Sv).

The PSAR states that since the end state and event consequences of the MTRBC DBE are essentially the same as the MTLPF DBE, the radionuclide release and key parameters are the same as presented in the PSAR section 3.4.2. The NRC staff finds that the end state and consequences from the event are similar to the MTLPF DBE and thus the radionuclide release and key parameters of the event reasonably represent the MTRBC DBE.

The MTRBC DBE is mapped to a MTLPF DBA as shown in PSAR table 3.2-4 since the event progression during a DBA and consequences of both the events are essentially similar. PSAR section 3.6.2 describes the MTLPF DBA.

Pressurized Loss of Forced Cooling non-DBA

PSAR section 3.4.7 describes the PLOFC analysis for non-DBA events and the long-term phase of several LBEs combined with the short-term transient phase to determine the dose consequence for a given event. The event assumes the HPB remains intact but there is no active cooling available and core cooling is provided primarily through passive RCCS and buoyancy-driven helium convection.

The PLOFC DBE event is modeled using the GOTHIC model for the Xe-100 reactor and is simulated for 30 days as the reactor transitions to a stable long-term plant response as described in PSAR section 3.1.3. The total dose consequence for the event is the sum of the doses from the short-term event modeled in Flownex and the long-term plant response modeled in GOTHIC, with both simulations starting the IE at time zero (i.e. the short-term plant response from Flownex is not an input to GOTHIC for long-term response). This modeling approach allows for an overall conservative dose estimate without modeling each LBE through its long-term evolution. A control rod withdrawal (SCRW) event was selected as a bounding representation of the long-term PLOFC conditions.

The PSAR states that the PLOFC case for non-DBA LBEs was analyzed using nominal values and notes that while the individual ESs for non-DBA events credit control rods, the PLOFC DBE evaluation does not model rod insertion as an added conservatism; however, the rods are credited to prevent recriticality. The NRC staff finds this acceptable since the assumption leads to a conservative result for dose consequences.

PSAR figures 3.4.7-1 through 3.4.7-5 provide the key parameters of HPB pressure and maximum and average fuel temperatures for the PLOFC event. The event shows that the HPB relief setpoint is not reached and there is no mass release of radionuclides. The PLOFC

contribution to the consequence is due to the normal HPB leakage and increase in fuel temperatures at the beginning of the SCRW event.

PSAR figure 3.4.7-2 shows maximum fuel temperatures for the nominal case briefly rise beyond 1600°C , where higher dose consequences are expected due to degradation of the SiC layer of the TRISO fuel observed in literature review (PNNL-31427, “TRISO Fuel: Properties and Failure Modes,” ML21175A152). The NRC staff confirmed in the regulatory audit that the EMs account for higher releases at these elevated temperatures, thus providing reasonable assurance that the preliminary event consequences for the PLOFC DBE are sufficient to assure safety margins during normal operations and transient conditions for the preliminary LMGS design. At the OL application stage, the NRC staff will confirm the accuracy of these models based on the planned testing to demonstrate performance of the Xe-100 fuel at these elevated temperatures as discussed in LMGS R&D program in PSAR chapter 13.

This event analyzes loss of forced cooling with no active heat removal available, but all PLOFC DBEs in PSAR table 3.2-2 end in an active cooling state, either through partial helium flow, SSS, feedwater cooling or active RCCS. Therefore, the analysis provided for these events is conservative relative to the ESs it represents. PSAR section 3.6.10 describes that the PLOFC DBE is mapped to the corresponding PLOFC DBA.

Depressurized Loss of Forced Cooling non-DBA

PSAR section 3.4.8 describes the DLOFC analysis for non-DBA events and the long-term phase of several LBEs with HPB break combined with the short-term transient phase to determine the dose consequence for a given event. The event assumes that: the HPB does not remain intact (i.e., assumes leaks or breaks); the reactor is depressurized to atmospheric levels; there is no active cooling available; and the core cooling is provided primarily through a passive RCCS.

Similar to the PLOFC DBE event, the DLOFC DBE is modeled using the GOTHIC model for the Xe-100 reactor and is simulated for 30 days as the reactor transitions to a stable long-term plant response as described in PSAR section 3.1.3. The total dose consequence for the event is conservatively assigned as the sum of the doses from the short-term event modeled in Flownex and the long-term plant response modeled in GOTHIC, with both simulations starting the IE at time zero.

The PSAR states that the primary release of radionuclides for the event is from a breach in the HPB. Additionally, small amounts of radioactive material can be released from the volumetric expansion of contaminated helium out of the HPB due to core heat up. Some leakage of contaminated helium out of the SST due to diurnal breathing is possible.

The DLOFC DBE evaluation is performed for two break sizes: 3 mm, which represents the lower threshold for leaks that the HSS cannot manage, and 65 mm, which corresponds to the largest pipe in the HPB. For each break size, six different DLOFC evaluations were performed by altering different PKPIRT parameters. The DLOFC DBE case presents results for the case with PKPIRT parameters set to nominal values.

The DLOFC DBE is initiated with a break in the primary HPB boundary, followed by depressurization to atmospheric levels. Following depressurization, the primary heat transfer mechanism is passive heat transfer through RCCS and conductive heat transfer from pebble

bed to graphite assembly. PSAR figure 3.4.8-1 shows that the 65 mm break size depressurizes the HPB in less than 10 minutes. This break size results in highest fuel temperatures. The DLOFC DBE evaluation does not model rod insertion for long-term events as added conservatism but does credit rods to prevent recriticality. The NRC staff find this acceptable since the assumption leads to a conservative result for dose consequences.

PSAR figures 3.4.8-1 through 3.4.8-5 provide the key parameters of HPB pressure and maximum and average fuel temperatures for the DLOFC event. The analysis shows that the peak fuel temperature of 1803°C is reached at approximately 20.3 hours for the nominal case with a 65 mm break size, while the nominal 3 mm break reaches the peak fuel temperature of 1685°C at 42.2 hours post break. The total 30-day EAB dose is calculated by adding dose from the DLOFC DBE combined with the short-term transient phase for the non-DBA events.

PSAR figure 3.4.8-2 shows that maximum fuel temperatures for the 65 mm break reach a peak of 1803°C and remain above 1600°C for about 4 days. Based on the NRC staff's literature review (PNNL-31427), higher dose consequences are expected beyond 1600°C, and 1800°C is generally characterized as the severe accident temperature. The NRC staff confirmed in the regulatory audit that the dose consequence models account for higher releases at these elevated temperatures, thus providing reasonable assurance that preliminary the event consequences for the DLOFC DBE are sufficient to assure safety margins during normal operations and transient conditions for the preliminary LMGS design. At the OL application stage, the NRC staff will confirm the accuracy of these models based on the planned testing to demonstrate performance of the Xe-100 fuel at these elevated temperatures as discussed in LMGS R&D program (PSAR chapter 13).

While for this event the applicant analyzed loss of forced cooling with no active heat removal available, all DLOFC DBEs in PSAR table 3.2-2 end in an active cooling state, either through SSS, feedwater cooling or active RCCS. Therefore, the NRC staff finds that the analyses provided for these events are conservative relative to the ESs. PSAR section 3.6.11 describes that the DLOFC DBE is mapped to corresponding DLOFC DBA.

Summary DBE Evaluations and Safety Margins

The NRC staff finds that the evaluation of DBEs in PSAR section 3.4 provides reasonable assurance of safety margins during DBEs because:

1. The applicant used an acceptable EM.

2. The applicant used suitably conservative inputs and boundary conditions for the evaluations.

3. The applicant describes the results of the evaluations including the event initiator, and the narrative of ES is sufficient to understand the plant response and identify the SSCs credited to mitigate the event.

4. The applicant provides the dose consequences and the details of the radionuclide release are provided where applicable, along with the key parameters of interest to demonstrate margin to the F-C curve.

The NRC staff will confirm the accuracy of consequence models at elevated temperatures at the OL application stage using information from the planned testing discussed in the LMGS R&D program in PSAR chapter 13.

3.4.1.2 Identification of SSCs necessary to prevent accidents and mitigate DBEs

Based on the event descriptions provided in PSAR section 3.4, the NRC staff identified the following SSCs credited to prevent and mitigate DBEs. Table 3.4-1 of this SE documents the results of the NRC staff's review of DBEs.

Table 3.4-1: SSCs Credited to Prevent and Mitigate DBEs

PSAR Section	DBE	Initiator	SSCs Credited to Prevent and Mitigate
3.4.1	MLOOP-23	Loss of Offsite Power	NIE, NIS, RPS, RCSS, RCCS
3.4.2	MTLPF-22	Loss of Primary Flow	NIE, NIS, DCS, RPS, RCSS, RCCS
3.4.3	MD-68	Medium HPB Depressurization	NIE, NIS, RPS, RCSS, RCCS
3.4.4	MFLB-01	Main Feedwater Line Break	NIE, NIS, RPS, RCSS, DCS, HCS, SSS
3.4.5	SSGTR-14	Steam Generator Tube Rupture	NIE, NIS, IPS, RPS, RCSS, DCS, RCCS
3.4.6	SCRW-13	Control Rod Withdrawal	NIE, NIS, RPS, RCSS, DCS, HCS, SSS
3.4.9	MTRBC-01	Loss of Reactor Building Cooling Water	NIE, NIS, IPS, RPS, RCSS, RCCS
3.4.7	PLOFC DBE	PLOFC Analysis for Non-DBA Events	NIE, NIS, IPS, RPS, RCSS[1], RCCS
3.4.8	DLOFC DBE	DLOFC Analysis for Non-DBA Events	NIE, NIS, IPS, RPS, RCSS[1], RCCS

1 The non-DBA PLOFC and DLOFC evaluations do not model RCSS insertion as an added conservatism, however, the short-term events associated with the non-DBA PLOFC and DLOFC evaluations credit rod insertion.

Based on the information provided in PSAR section 3.4, the NRC staff finds that the application clearly identified the SSCs credited to prevent and mitigate DBEs in the event descriptions.

3.4.1.3 Uncertainty Associated with Performance of SSCs Necessary to Prevent and Mitigate DBE Consequences

RG 1.253, section C.4, "Licensing Basis Events," describes the connection between evaluating uncertainty associated with performance of SSCs and the identification of safety features and components that require R&D as follows:

> [These criteria] support[s] staff findings associated with 10 CFR 50.35(a)(3), namely, that the application describes the safety features and components that require research and development, and that the applicant will conduct a

reasonably designed research and development program to resolve any associated safety questions.

PSAR table 1.4-11 identifies the scope of work needed to address L&Cs associated with the use of the TSAM TR which includes uncertainty quantifications. Further, an evaluation of operating band for the input parameters used in the LBE evaluation and treatment of uncertainties is needed to ensure the LBE evaluations yield conservative results for the 95th percentile consequence is presented in SE section 3.1.1 above.

The applicant provided a commitment to complete uncertainty quantification at the OL application stage. This provides reasonable assurance that any potential safety questions associated with the adequacy of SSCs and the integral system relied for mitigation of DBEs will be addressed prior to the issuance of an OL.

3.4.2 Conclusion

Based on its review, the NRC staff finds that the description of DBEs provided in PSAR section 3.4, is consistent with the applicable regulatory guidance and meets the regulatory requirements identified in 10 CFR 50.34(a)(1), 10 CFR 50.34(a)(4) and 10 CFR 50.35(a)(3). This finding is based on the NRC staff's conclusions that the design will provide sufficient safety margins during normal operations and transient conditions, the applicant has identified the SSCs necessary to prevent accidents and mitigate DBEs, and the applicant has demonstrated an understanding of the uncertainty associated with the performance of SSCs necessary to prevent accidents and mitigate DBE consequences.

3.5 Beyond-Design-Basis Events

The NRC staff reviewed PSAR section 3.4, "Beyond Design Basis Events," which presents the evaluation of representative or bounding BDBEs for the LMGS. Table 3.2-3 of this SE lists the BDBEs presented in PSAR sections 3.5 with their PRA ES identification and event initiator.

Also, PSAR section 3.2 discusses LBEs which are not explicitly evaluated in PSAR section 3.5 as BDBEs. While some of the LBEs discussed in PSAR section 3.2 are represented by LBEs with similar ESs; other LBEs, as discussed in PSAR table 3.2-7, do not have an explicit evaluation performed in the PSAR, which includes the BDBEs. In PSAR section 3.2, the applicant committed to performing a fully refined LBE evaluation and a final ES family grouping in support of a future OL application. The NRC staff notes that the PRA methodology for the BDBE selection is limited to preliminary design for full-power operations at the CP application stage. As the PRA is expanded beyond full power operation core events, the events analyzed and their outcomes may lead to a different set of LBEs to be identified as BDBEs at the OL application stage.

The NRC staff's review and acceptability of the identification of representative or bounding BDBEs is presented in section 3.2 of this SE.

NEI 21-07 defines high-consequence BDBEs as those with consequences that exceed 10 CFR 50.34 dose limits. Consistent with NEI 21-07, control room doses are not reported for BDBEs.

The applicable regulatory requirements for the evaluation of the BDBEs are as follows:

- 10 CFR 50.34(a)(1) and (4); and
- 10 CFR 50.35(a)(3).

The applicable guidance for the evaluation of the LBE summary are as follows:

- RG 1.233, Revision 0, which endorses NEI 18-04, Revision 1, with additions; and
- RG 1.253, Revision 0, which endorses NEI 21-07, Revision 1, with clarifications and additions.

3.5.1 Technical Evaluation

The NRC staff evaluated the information in PSAR section 3.5 against the following criteria for LBE evaluations in a CP application provided in RG 1.253, section C.4:

1. The design will provide sufficient safety margins during normal operations and transient conditions (evaluated in section 3.5.1.1 of this SE).
2. The applicant has identified the SSCs necessary to prevent accidents and mitigate accident consequences (evaluated in section 3.5.1.2 of this SE).
3. The applicant has demonstrated an understanding of the uncertainty associated with the performance of SSCs necessary to prevent accidents and mitigate accident consequences (evaluated in section 3.5.1.3 of this SE).

3.5.1.1 Evaluation of safety margins during normal operation and transient conditions

The NRC staff evaluated the assessment of safety margins during BDBEs by reviewing the EM, input parameters and initial conditions, and analysis results.

Evaluation Model

The EM for evaluation of the BDBEs is identical to the EM described in section 3.3 of this SE for the AOOs. Therefore, the conclusion on the EM in section 3.3 for the AOOs is also applicable for BDBEs. The EM for evaluation of the BDBEs is identical to the EM described in section 3.3 of this SE for the AOOs. Therefore, the conclusion on the EM in section 3.3 for the AOOs is also applicable for the BDBEs.

Input Parameters and Initial Conditions (General)

The general input parameters and initial conditions in PSAR table 3.3-1 for evaluation of the BDBEs are identical to the input parameters and initial conditions described in section 3.3 of this SE for the AOOs. Therefore, the conclusion on the general input parameters and initial conditions in section 3.3 for the AOOs is also applicable for the BDBEs.

BDBE Specific Plant Conditions and Results

The NRC staff's evaluation of the BDBE specific plant conditions and results are presented below. For each BDBE, the NRC staff ensured that, consistent with RG 1.233: (1) the event initiator and the ES narrative in the PSAR is sufficient to understand the plant response and identify the SSCs credited to mitigate the event, and (2) the dose consequences and the details of the radionuclide release are provided, along with the key parameters of interest, and demonstrate margin to the F-C curve.

The NRC staff notes that the ES description and the end state in PSAR section 3.5 may be different than the actual plant response provided for the BDBE in PSAR table 3.2-2. This is attributed to LME describing the consequence analysis sequence in PSAR section 3.5, rather than the ES. In many cases, LME applied conservative assumptions that would represent additional failures which are reflected by the differences in the ES descriptions between PSAR table 3.2-2 and section 3.5. The NRC staff finds this acceptable since the ES descriptions in PSAR table 3.2-2 were confirmed to be accurate and LME confirmed in the audit that the differences were due to the conservatism used in the event modeling.

The NRC staff's review of each BDBE evaluation is documented below.

Control Rod Group Withdrawal BDBE

PSAR section 3.5.1 describes the CRW BDBE analysis and the events composing PRA sequence SCRW-03. Of the full scope of control rod withdrawal BDBEs, this event has the bounding cooling mode but involves only one module. There are four other control rod withdrawal BDBEs that involve all four modules with similar sequences descriptions but with doses that are four times higher. Control rod withdrawal BDBEs that do not end in active RCCS cooling do not have a release based on the mean consequence analysis.

The PSAR states that the ES for the SCRW-03 BDBE is similar to the ES for the SCRW DBE except that SSS is not available and long-term cooling is provided by the active RCCS. The IE for this BDBE is withdrawal of a full control rod bank at full power at rate of 1.2 m/sec. The PSAR states that the withdrawal rate is 20 percent faster than system design. Following event initiation, RPS initiates an RT on high neutron flux, resulting in the insertion of control rods and shutdown rods. The helium circulators subsequently trip on high He/FW ratio and DCS fails to initiate cooling on SSS. With main loop and SSS both failing, the active RCCS provides long-term cooling. The event ends with the reactor shutdown to decay power level with intact HPB and long-term cooling (PLOFC) provided by active RCCS. The consequence analysis sequence for this event in PSAR section 3.5 models long-term PLOFC cooling with passive RCCS. The NRC staff finds this to be acceptable since this conservative assumption represents additional failures and leads to bounding dose consequences.

The SCRW BDBE has a total mean and 95th percentile 30-day radiation dose (TEDE), including the long-term cooldown, greater than 2.5 mrem (0.025 mSv). However, the mean and 95th percentile dose are below the 25-rem (0.25 Sv) limit for the BDBEs. PSAR table 3.5.1-1 provides details of the radionuclide release for the event, including the long-term PLOFC event. PSAR figures 3.5.1-1 through 3.5.1-8 provide the details of key parameters for this BDBE including control rod position, neutron flux, reactor power, reactivity, HPB pressure and fuel temperatures.

Small HPB Depressurization BDBE

PSAR section 3.5.2 describes the SD BDBE analysis and the events composing PRA sequence SD-74. This section also describes the most limiting PRA ES, SD-131.

The PSAR states that the ES and consequences for SD-74 are similar to the ES and consequences for the SD AOO discussed in PSAR section 3.3.5, except that the forced circulation ends earlier and results in higher fuel temperatures and therefore higher offsite dose. The IE for this BDBE is a 5 mm break in the reactor inlet cross-vessel duct. The PSAR states that three ranges of breaks between 1 mm and 10 mm were chosen for the evaluation and the 5 mm break generally represents the event. The sensitivity studies for the event included a 10 mm break. Following the event initiation, the HPB starts to depressurize, and RPS initiates an RT on the low pressure set point, resulting in the insertion of control rods and shutdown rods. The helium circulators subsequently trip on RPS high mass flow ratio and long-term DLOFC cooling is provided by RCCS. PSAR figure 3.5.2-3 shows that the leak continues for several hours until atmospheric pressure is reached.

The PSAR details a sensitivity study performed on break size and location. PSAR table 3.5.2-1 shows the results of the sensitivity study on the break sizes and break location and includes the status of active systems for the event.

The PSAR also provides the sequence of events for the SD-131 event where the reactor fails to trip with unsuccessful primary system pumpdown in all four modules and active RCCS provides long-term cooling. The PSAR discussion states that the event progresses to successful passive RCCS cooling and subsequently successful active RCCS cooling. The NRC staff confirmed, during the regulatory audit that the long-term cooling is provided by active RCCS. The offsite dose for this event is higher than other SD BDBEs identified due to the trip failure experienced by all four modules. The consequence analysis sequence for this event in PSAR section 3.5 models long-term PLOFC cooling with passive RCCS. The NRC staff finds this to be acceptable since this conservative assumption represents additional failures and leads to bounding dose consequences.

The SD BDBEs (SD-74 and SD 131) have a total mean and 95th percentile 30-day radiation dose (TEDE), including the long-term cooldown, greater than 2.5 mrem (0.025 mSv). PSAR table 3.2-3 shows the total multi-module mean for the SD-131 event as 9.16 rem (0.0916 Sv) with a 95th percentile dose of 10.2 rem (0.102 Sv). The mean and 95th percentile doses for the event are below the 25-rem (0.25 Sv) limit for BDBEs. Details of the radionuclide release for both the events, including the long-term PLOFC event, are provided in PSAR tables 3.5.2-2 and 3.5.2-3. PSAR figures 3.5.2-1 through 3.5.2-4 describe the details of key parameters for SD-74 BDBE, including sensitivity cases, for reactivity, fuel temperature, mass flow and circulator flow rates.

The NRC staff notes that PSAR figure 3.5.2-2 shows that maximum fuel temperatures for the SD-74 sensitivity cases reach above 1600°C, where higher dose consequences are expected due to degradation of the SiC layer of the TRISO fuel, as observed in literature review (PNNL-31427). The NRC staff confirmed in the regulatory audit that the dose consequence models account for higher releases at these elevated temperatures, thus providing reasonable assurance at the preliminary stage of the LMGS design, that the dose consequences for SD BDBE show margin to F-C curve during normal operations and transient conditions. At the OL application stage, the NRC staff will confirm the accuracy of these models based on the planned

testing to demonstrate performance of the Xe-100 fuel at these elevated temperatures as discussed in LMGS R&D program in PSAR chapter 13.

Medium HPB Depressurization BDBE

PSAR section 3.5.3 describes the MD BDBE analysis and the events composing PRA sequence MD-65.

The PSAR states that the ES and consequences are similar to the ES and consequences for the MD DBE discussed in PSAR section 3.4.3, except that SSS fails for the BDBE and long-term cooling is performed actively by the RCCS. The IE for this BDBE is a 65 mm break in the refueling tube at the top of the reactor which corresponds to largest pipe in the HPB. The PSAR states that three break sizes; 10 mm, 25 mm, and 65 mm were chosen for the evaluation, where the 25 mm break size represents the geometric mean of the other two break sizes.

Following the event initiation, the HPB rapidly depressurizes, and RPS initiates an RT on low pressure setpoint, resulting in the insertion of control rods and shutdown rods. The helium circulators subsequently trip on RPS high mass flow ratio and active RCCS provides long-term DLOFC cooling. The consequence analysis sequence for this event in PSAR section 3.5 models long-term DLOFC cooling with passive RCCS. The NRC staff finds this to be acceptable since this conservative assumption represents additional failures and leads to bounding dose consequences.

The PSAR details a sensitivity study performed on break size and location and results for key parameters are provided for all of the cases: 65 mm break at the reactor inlet with SSS cooling, 65mm and 25 mm break at reactor inlet with active RCCS cooling, and a 65 mm break at the defueling chute with active RCCS cooling. PSAR figure 3.5.3-5 shows that maximum fuel temperatures are higher for a 25 mm break during the transition period, but the overall maximum fuel temperatures are bounded by the DFLOC DBE detailed in PSAR section 3.4.8.

The MD BDBE has a total mean and 95th percentile 30-day radiation dose (TEDE), including the long-term cooldown, greater than 2.5 mrem (0.025 mSv). However, the mean and 95th percentile doses are below the 25-rem (0.25 Sv) limit for the BDBEs. PSAR table 3.5.3-1 provides details of the radionuclide release for the event, including the long-term DLOFC event. PSAR figures 3.5.3-1 through 3.5.3-5 describe the details of key parameters reactor inlet pressure, power, reactivity, circulator speed and maximum fuel temperatures for the different sensitivity studies on break size and location.

Large HPB Depressurization BDBE

PSAR section 3.5.4 describes the LD BDBE analysis and the events composing PRA sequence LD-01.

The IE for this BDBE is a 75 mm break at the reactor inlet which bounds the largest pipe in the HPB. The PSAR states that three break sizes; 75 mm, 98.7 mm, and 150 mm, were chosen for the evaluation, where 98.7 mm represents the geometric mean of the other two break sizes. Following the event initiation, the HPB rapidly depressurizes, and RPS initiates an RT on the low pressure setpoint, resulting in the insertion of control rods and shutdown rods. The helium circulators subsequently trip on RPS high mass flow ratio and passive RCCS provides long-term DLOFC cooling. The event shows that the fuel temperatures continue to rise at end of the short-

term event and then stabilize during the DLOFC cooling phase, as described in PSAR section 3.4.8.

The LD BDBE has a total mean and 95th percentile 30-day radiation dose (TEDE), including the long-term cooldown, greater than 2.5 mrem (0.025 mSv). However, the mean and 95th percentile dose is below the 25-rem (0.25 Sv) limit for BDBEs. PSAR table 3.5.4-1 provides details of the radionuclide release for the event, including the long-term PLOFC event. PSAR figures 3.5.4-1 and 3.5.4-2 describe the details of key parameters of the HPB pressure and core temperatures.

The NRC staff notes that the maximum and average fuel temperature presented in PSAR figure 3.5.4-2 are still rising at the end of the short-term event. However, the results for fuel temperatures presented for the DLOFC case in PSAR section 3.4.8, which represents the long-term evaluation for this BDBE, bound the fuel temperatures for this event. Therefore, the NRC staff finds the fuel temperature results to be acceptable.

Loss of Steam Generator Feedwater BDBE

PSAR 3.5.5 describes the LOFW BDBE analysis and the events composing PRA sequence MTLFW-03.

The IE for this BDBE is a total loss of main feedwater to both SG bundles due to tripping of the HPFWP motors in one reactor module. The loss of feedwater flow results in the loss of heat transfer capability from the primary to secondary side which leads to an RPS trip and rod insertion. Immediately after, the circulators trip on high He/FW flow ratio. The HPB relief set point is not reached for the event. The analysis assumes failure of the feedwater pumps and the SSS system. The event ends in a safe, stable shutdown with passive cooling in PLOFC mode provided via the RCCS. PSAR section 4.7 describes the PLOFC event.

The MTLFW BDBE has a total mean and 95th percentile 30-day radiation dose (TEDE), including the long-term cooldown, greater than 2.5 mrem (0.025 mSv). However, the mean and 95th percentile dose is below the 25-rem (0.25 Sv) limit for the BDBE. PSAR table 3.5.5-1 provides details of the radionuclide release for the event, including the long-term PLOFC event. PSAR figures 3.5.5-1 through 3.5.5-4 describe the details of key parameters of reactor power, fuel temperature, feedwater flow and HPB pressure.

The NRC staff notes that the maximum fuel temperature presented in PSAR figure 3.5.5-2 is still rising at the end of the short-term event. However, the results for fuel temperatures presented for the DLOFC case in PSAR section 3.4.8, which represents the long-term evaluation for this BDBE, bound the fuel temperatures for this event. Therefore, the NRC staff finds the fuel temperature results to be acceptable.

PSAR section 3.6.5 maps the MTLFW-03 BDBE to the MTLFW DBA instead of the MTLFW DBE. The PSAR states that this mapping was done due to the end state for the events indicating that the offsite dose for the MTLFW DBE would be the same as the offsite dose for a BDBE. The NRC staff's review and acceptability of the DBA mapping is presented in section 3.6.1 of this SE.

Main Feedwater Line Break BDBE

PSAR section 3.5.6 describes the MFLB BDBE analysis and the events composing PRA sequence MFLB-02.

The PSAR states that the ES and consequences are similar to the ES and consequences for the MFLB DBE discussed in PSAR section 3.4.4, except that SSS fails for the BDBE and active RCCS provides long-term cooling.

The IE for this DBE is a double guillotine break of the combined feed line header. RPS initiates an RT on high cold helium temperature at the SG outlet and results in the insertion of control rods and shutdown rods. Following rod insertion, the helium circulators also trip on high cold helium temperature at the SG outlet. Following the trip, the SSS operation fails. The event ends in a safe, stable shutdown with active cooling in PLOFC mode provided via the RCCS. The consequence analysis sequence for this event in PSAR section 3.5 models long-term PLOFC cooling with passive RCCS. The NRC staff finds this to be acceptable since this conservative assumption represents additional failures and leads to bounding dose consequences.

The MFLB BDBE has a total mean and 95th percentile 30-day radiation dose (TEDE), including the long-term cooldown, greater than 2.5 mrem (0.025 mSv). However, the mean and 95th percentile dose is below the 25-rem (0.25 Sv) limit for BDBEs. PSAR table 3.5.6-1 provides details of the radionuclide release for the event, including the long-term PLOFC event. PSAR figure 3.5.6-1 provides the maximum and average fuel temperatures as well as the moderator and reflector temperatures for the event.

The NRC staff notes that the maximum fuel temperature presented in PSAR figure 3.5.6-1 is still rising at the end of the short-term event. However, the results for fuel temperatures presented for the PLOFC case in PSAR section 3.4.7, which represents the long-term evaluation for this BDBE, bound the fuel temperatures for this event. Therefore, the NRC staff finds the fuel temperature results to be acceptable.

Main Steamline Break BDBE

PSAR section 3.5.7 describes the MSLB BDBE analysis and the events composing PRA sequence MSLB-08.

The PSAR states that the ES for MSLB-08 is identical to the ES for the MSLB AOO discussed in PSAR section 3.3.2 except that the SSS cooling is unavailable for the BDBE and the long-term cooling is provided by the RCCS in the PLOFC mode.

The IE for this BDBE is a guillotine break downstream of the tee joining the supply lines from two separate SG tubes. A break at this location maximizes the rupture size and affects both tube bundles. Following event initiation, RPS and IPS initiate an RT on high helium to feedwater flow ratio, resulting in the insertion of control rods and shutdown rods and tripping the helium circulators. Following the trip, SSS operation fails and cooling is provided by active RCCS. The event ends in a safe, stable shutdown with active cooling in PLOFC mode provided via the RCCS. The consequence analysis sequence for this event in PSAR section 3.4 models long-term PLOFC cooling with passive RCCS. The NRC staff finds this to be acceptable since this conservative assumption represents additional failures and leads to bounding dose consequences.

The MSLB BDBE has a total mean and 95th percentile 30-day radiation dose (TEDE), including the long-term cooldown, greater than 2.5 mrem (0.025 mSv). However, the mean and 95th percentile dose is below the 25-rem (0.25 Sv) limit for BDBEs. PSAR table 3.5.7-1 provides details of the radionuclide release for the event, including the long-term PLOFC event. PSAR figures 3.5.7-1 through 3.5.7-8 describe the key parameters of main steam temperature, main steam pressure, rod position, reactivity, transient power, fuel temperature and HPB pressure.

Steam Generator Tube Leak BDBE

PSAR section 3.5.8 describes the SGTL BDBE analysis and the events composing PRA sequence SSGTL-21.

The PSAR states that the ES and consequences for SSGTL-21 are similar to the ES and consequences for the SSGTL AOO discussed in PSAR section 3.3.7 up until the SG isolation is complete. The IE for this BDBE is a 3.16 mm break in SG tube bundle A. The break size represents the geometric mean of the defined SSGTL range of 1 mm to 10 mm diameter. The RPS initiates an RT on high moisture and results in the insertion of control rods and shutdown rods. Following this, the RPS initiates SG isolation. The water in the leaking SG bundle slowly leaks into the HPB through the break until the pressure on both sides equalizes. The HPB relief valve opens and closes once during the event and remains closed. The event ends in a safe, stable shutdown with active cooling in PLOFC mode provided via the RCCS. The consequence analysis sequence for this event in PSAR section 3.5 models long-term PLOFC cooling with passive RCCS. The NRC staff finds this to be acceptable since this conservative assumption represents additional failures and leads to bounding dose consequences.

PSAR figure 3.5.8-3 shows that the helium circulator speed does not go below 500 rpm even after the trip and states that this is an artifact of circulator friction model not being included in the Flownex code. The helium flow ceases when the circulators trip. The PSAR evaluation presented for the BDBE also lists the SSGTL-06 event as the most limiting BDBE of this event series since this event assumes lifting of HPB safety valves on all four modules resulting in higher releases as well as failure of the SSS and a degraded RCCS, with failure to shut down the unaffected modules.

The degraded RCCS for the limiting SSGTL BDBE event results in multiple HPB relief valve lifts. The event leads to higher dose consequences. Since no specific analysis is available for the event in the PSAR, the dose assignment for this event is assigned to a seismic-B DBA as detailed in PSAR section 3.6.12. The PSAR states that this is conservative since the seismic-B DBA impacts all four modules as opposed to a single module for the SSGTL-06 event. The PSAR provides a commitment to demonstrate the associated safety analysis for the event at the OL application stage to show that degraded RCCS cooling will still provide sufficient heat transfer to keep the HPB intact.

The NRC staff finds this acceptable at the CP application stage since the dose consequence of the event is assigned to a bounding DBA. The NRC staff would review the detailed evaluation when submitted as part of an OL application.

PSAR table 3.5.8-1 provides details of the radionuclide release for the SSGTL-21 event, including the long-term PLOFC event. PSAR figures 3.5.8-1 through 3.5.8-9 describe the key parameters of humidity, rod position, circulator speed, feedwater and steam isolation valve position, HPB and SG outlet pressure, tube rupture flow and reactivity for the event.

Steam Generator Tube Rupture BDBE

PSAR section 3.5.9 describes the SGTR BDBE analysis and the events composing PRA sequence SSGTR-29.

The SSGTR BDBE IE is a 15.8 mm break near the top of one of the SG bundles. This value represents the geometric mean of the defined SSGTR range of 10 mm to 25 mm. Following the event initiation, the HPB moisture signals from the IPS and RPS result in an RT leading to insertion of control and shutdown rods followed by a circulator trip. SG isolation is assumed to fail, resulting in an unisolated SG for the transient. The DCS controller responds to the dropping pressure by closing the valve and greatly restricting the feedwater flow.

The isolation of the affected SG has two steps: feedwater-side isolation and steam-side isolation. Following the opening and subsequent closing of the SG dump valves, the DCS initiation of the SSS cooling fails. The HPB fully depressurizes during the event, and the entire circulating inventory is assumed to be released at the end of the event such that any holdup of primary side activity in the secondary side is not credited. The event ends in a safe, stable shutdown condition with active cooling in DLOFC mode provided via the RCCS. The consequence analysis sequence for this event in PSAR section 3.4 models long-term DLOFC cooling with passive RCCS. The NRC staff finds this to be acceptable since this conservative assumption represents additional failures and leads to bounding dose consequences.

The SSGTR BDBE has a total mean and 95th percentile 30-day radiation dose (TEDE), including the long-term cooldown, greater than 2.5 mrem (0.025 mSv). However, the mean and 95th percentile dose is below the 25-rem (0.25 Sv) limit for BDBEs. PSAR table 3.5.9-1 provides details of the radionuclide release for the event, including the long-term DLOFC event. PSAR figures 3.5.9-1 through 3.5.9-8 describe the key parameters of humidity, rod position, reactivity, SG steam outlet temperature, turbine mass flow, tube rupture flow, fuel temperature and HPB pressure for the event.

Loss of Primary Flow BDBE

PSAR section 3.5.10 describes the loss of primary flow (LOPF) BDBE and the events composing PRA sequence STLPF-25.

The IE for the STLPF BDBE is loss of primary flow for both circulators from an initial full power state due to a potential spurious trip signal from DCS, IPS, or RPS. Following the IE, the circulators begin to coastdown, DCS reduces the feedwater flow to match the primary flow, RPS trips the reactor on high mass flow ratio, and control and shutdown rods insert. The circulator restoration is unsuccessful. Modules 2 through 4 also trip on IPS and RPS setpoints due to failure to maintain system balance and degraded RCCS cools the plant passively.

The applicant does not provide details of radionuclide release and key parameters for the event. The PSAR states that the STLPF DBE is assigned the seismic-B DBA dose since no specific event is available at the CP stage to model the degraded RCCS performance. The PSAR provides a commitment to demonstrate associated safety analysis for the event at the OL application stage to show that degraded RCCS cooling will still provide sufficient heat transfer to keep the HPB intact. The NRC staff finds this acceptable at the CP stage since the dose consequence of the event is assigned to a bounding DBA. The NRC staff would review the detailed evaluation when submitted as part of a future OL application.

Summary BDBE Evaluations and Safety Margins

The NRC staff finds that the evaluation of BDBEs in PSAR section 3.5 provides reasonable assurance of safety margins during BDBEs because:

1. The applicant used an acceptable EM.

2. The applicant used suitably conservative inputs and boundary conditions for the evaluations.

3. The applicant describes the results of the evaluations including the event initiator, and the narrative of ES is sufficient to understand the plant response and identify the SSCs credited to mitigate the event.

4. The applicant provides the dose consequences, and the details of the radionuclide release are provided where applicable, along with the key parameters of interest to demonstrate margin to the F-C curve.

The NRC staff notes that for some of the BDBEs where no analysis is available in the PSAR, the dose assignment for those events is assigned to a seismic-B DBA. The NRC staff finds this acceptable at the CP stage based on the dose consequence for the event being assigned to a higher consequence event, and the applicant's commitment to evaluate those events at the OL application stage.

The PSAR does not identify any high-consequence BDBEs, i.e., those with consequences that exceed 25 rem (0.25 Sv) based on the preliminary design and limited scope of the PRA and safety analysis.

3.5.1.2 Identification of SSCs necessary to prevent accidents and mitigate BDBEs

Based on the event descriptions provided in PSAR section 3.5, the NRC staff identified the following SSCs credited to prevent accidents and mitigate DBEs. Table 3.5-1 of this SE documents the results of the NRC staff's review of the BDBEs.

Table 3.5-1: SSCs Credited to Prevent and Mitigate BDBEs

PSAR Section	DBE	Initiator	SSCs Credited to Prevent and Mitigate
3.5.1	SCRW-03	Control Rod Withdrawal	NIE, NIS, RPS, RCSS, RCCS
3.5.2	SD-131	Small HPB Depressurization	NIE, NIS, RPS, RCSS, RCCS
3.5.3	MD-65	Medium HPB Depressurization	NIE, NIS, RPS, RCSS, RCCS
3.5.4	LD-01	Large HPB Depressurization	NIE, NIS, RPS, RCSS, RCCS
3.5.5	MTLFW-03	Loss of Steam Generator Feedwater	NIE, NIS, RPS, RCSS, RCCS

3.5.6	MFLB-02	Main Feedwater Line Break	NIE, NIS, RPS, RCSS, RCCS
3.5.7	MSLB-08	Main Steam Line Break	NIE, NIS, IPS, RPS, RCSS, RCCS
3.5.8	SSGTL-21	Steam Generator Tube Leak	NIE, NIS, RPS, RCSS, RCCS
3.5.9	SSGTR-29	Steam Generator Tube Rupture	NIE, NIS, RPS, RCSS, RCCS
3.5.10	STLPF-25	Loss of Primary Flow	NIE, NIS, DCS, RPS, RCSS, RCCS

Based on the information provided in PSAR section 3.5, the NRC staff finds that sufficient information is provided to identify SSCs credited to prevent and mitigate BDBEs because the application clearly identified the SSCs credited to prevent and mitigate BDBEs using the information provided in the event descriptions

3.5.1.3 Uncertainty Associated with Performance of SSCs Necessary to Prevent and Mitigate BDBE Consequences

RG 1.253, section C.4, “Licensing Basis Events,” describes the connection between evaluating uncertainty associated with performance of SSCs and the identification of safety features and components that require R&D as follows:

> [These criteria] support[s] staff findings associated with 10 CFR 50.35(a)(3), namely, that the application describes the safety features and components that require research and development, and that the applicant will conduct a reasonably designed research and development program to resolve any associated safety questions.

PSAR table 1.4-11 identifies the scope of work needed to address L&Cs associated with the use of the TSAM TR which includes uncertainty quantifications. Further, an evaluation of the operating band for the input parameters used in LBE evaluations and the treatment of uncertainties needed to ensure LBE evaluations yield conservative results for 95th percentile consequence is presented in SE section 3.1.1 above.

The applicant provided a commitment to complete uncertainty quantification at the OL application stage. This commitment provides reasonable assurance that any potential safety questions associated with the adequacy of SSCs and the integral system relied on for mitigation of the BDBEs will be addressed prior to the issuance of an OL.

3.5.2 Conclusion

Based on its review, the NRC staff finds that the BDBE information described in PSAR section 3.5 is consistent with the applicable regulatory guidance and meets the regulatory requirements identified in 10 CFR 50.34(a)(1), 10 CFR 50.34(a)(4) and 10 CFR 50.35(a)(3). This finding is based on the NRC staff’s conclusions that the design will provide sufficient safety margins during normal operations and transient conditions, the applicant has identified the SSCs necessary to prevent and mitigate BDBEs, and the applicant has demonstrated an understanding of the uncertainty associated with the performance of SSCs necessary to prevent and mitigate BDBE consequences. (See section 3.5.1.3 of this SE).

3.6 Design-Basis Accidents

The NRC staff reviewed PSAR section 3.6, which presents the deterministic evaluation of transients derived from the analyses of DBEs for the LMGS in accordance with NEI 18-04. Table 3.2-5 of this SE presents DBAs evaluated in the PSAR along with associated mapped LBEs.

The DBA analysis is performed with only SR SSCs credited. The PSAR states that the DBA analysis performed does not model insertion of the SR control rods for the short-term transient as an added conservatism. Further, for the long-term PLOFC and DLOFC events, the ES does not model rod insertion; however, the rods are credited to prevent recriticality. The NRC staff finds this acceptable because the assumption leads to a conservative result for dose consequences.

PSAR section 3.2 discusses LBEs which are not explicitly evaluated in PSAR sections 3.3, 3.4, and 3.5. While some of these events are represented by LBEs with similar sequences, other LBEs are not explicitly evaluated in the PSAR. PSAR table 3.2-7 presents a discussion of these LBEs that are not explicitly evaluated anywhere in the PSAR and are not represented by other similar LBEs. These events include DBEs and other LBEs, which did not get mapped into corresponding DBAs. In PSAR section 3.2, the applicant committed to performing a fully refined LBE evaluation and a final ES family grouping in support of a future OL application.

The NRC staff notes that the PRA methodology for LBE selection is based on the preliminary design for full-power operations at the CP application stage. As the PRA is expanded beyond full power operation core events, the events analyzed and their outcomes may change.

The applicable regulatory requirements for the evaluation of the DBAs are as follows:

- 10 CFR 50.34(a)(1) and (4);
- 10 CFR 50.34(g);
- 10 CFR 50.35(a); and
- 10 CFR 50.44(d).

The applicable guidance for the evaluation of the DBAs is as follows:

- RG 1.233, Revision 0, which endorses NEI 18-04, Revision 1, with additions; and
- RG 1.253, Revision 0, which endorses NEI 21-07, Revision 1, with clarifications and additions

3.6.1 Technical Evaluation

The NRC staff evaluated the information in PSAR section 3.6 against the following criteria for LBE evaluations for a CP application provided in RG 1.253, section C.4:

1. The design will provide sufficient safety margins during normal operations and transient conditions (evaluated in section 3.6.1.1 of this SE).

2. The applicant has identified the SSCs necessary to prevent accidents and mitigate accident consequences (evaluated in section 3.6.1.2 of this SE).

3. The applicant has demonstrated an understanding of the uncertainty associated with the performance of SSCs necessary to prevent accidents and mitigate accident consequences (evaluated in section 3.6.1.3 of this SE).

3.6.1.1 Evaluation of safety margins during normal operations and transient conditions

The NRC staff evaluated the assessment of safety margins during DBAs by reviewing the EM, input parameters and initial conditions, and analysis results.

Evaluation Model

The EM for the evaluation of the DBAs is identical to the EM described in section 3.3 of this SE for the AOOs. Therefore, the conclusion on the EM in section 3.3 for AOOs is also applicable for the DBAs.

Input Parameters and Initial Conditions (General)

The general input parameters and initial conditions in PSAR table 3.3-1 for the evaluation of the DBAs are identical to the input parameters and initial conditions described in section 3.3 of this SE for the AOOs. Therefore, the conclusion on the general input parameters and initial conditions in section 3.3 for AOOs is also applicable for the DBAs.

The primary differences between DBA and non-DBA evaluations, as identified in PSAR table 3.2-1, are that for DBAs only SR SSCs are credited for mitigation, input parameters biased based on PKPIRT analysis, control rods not credited for an RT, and conservative values are used for the atmospheric dispersion factor and fuel failure fraction. The table also indicates that conservative or 95^{th} percentile values were used for steam, wash off, and liftoff contributions to radionuclide release. PSAR section 3.6 lists parameters for which appropriate uncertainties from PKPIRT analysis were applied to the DBA analysis.

The NRC staff finds the EM differences to be acceptable since the changes, as compared to non-DBA events, lead to more conservative plant parameters and dose consequences in general.

DBA Specific Plant Conditions and Results

The NRC staff's evaluation of the DBA specific plant conditions and results are presented below. For each DBA, the NRC staff ensured that, consistent with RG 1.233: (1) the event initiator and the ES narrative in the PSAR is sufficient to understand the plant response and

identify the SR SSCs credited to mitigate the event, and (2) the dose consequences and the details of the radionuclide release are *provided*, along with the key parameters of interest.

The NRC staff notes that the ES description and the end state in PSAR section 3.6 may be different than the actual plant response provided for the events they are mapped from in PSAR table 3.2-2. This is attributed to LME describing the consequence analysis sequence in PSAR section 3.6, rather than the ES. In many cases, LME applied conservative assumptions that would represent additional failures which are reflected by the differences in the ES descriptions between PSAR table 3.2-2 and section 3.6. The NRC staff finds this acceptable since the ES descriptions in PSAR table 3.2-2 were confirmed to be accurate and LME confirmed in the audit that the differences were due to the conservatism used in the event modeling.

The NRC staff's review of each DBE evaluation is documented below.

Control Rod Group Withdrawal DBA

PSAR section 3.6.1 describes the CRW DBA analysis and the events composing the PRA sequence mapped from the SCRW DBE with only SR systems credited and rods failing to insert.

The IE for this DBA is an unintended full bank control rod withdrawal from an initial inserted depth of ~183 cm below the bottom of the top reflector at a rate 20 percent faster than the system design. Following the IE, the control and shutdown rods fail to insert and RPS trips the circulators based on the failed RT. Both the main loop forced cooling and SSS are assumed to fail. The reactor shuts down on inherent reactivity feedback. The event ends with the reactor shutdown to decay power level with intact HPB and long-term cooling (PLOFC) provided by passive RCCS.

The NRC staff reviewed the event description in the PSAR which shows that the reactor power during the event reaches a maximum of approximately 480 MW. The rapid increase in power leads to an increase in system pressure but the system pressure remains below the 7 MPa design limit. The positive reactivity insertion due to rod withdrawal is countered with inherent negative reactivity, which drives the power below 1 MW approximately 400 seconds into the event.

The SCRW DBA results in a total radiation dose (TEDE) at the EAB of less than 1 rem, below the 25-rem (0.25 Sv) limit for DBAs. PSAR table 3.6-1 provides details of the radionuclide release for the event, including the long-term PLOFC event. The transient portion of the SCRW DBA results in a slight increase in the radiation dose at the EAB due to the heat up fuel, and the primary contributor to the radiation dose is the normal HPB leakage from circulating radioactive inventory during the long-term PLOFC. PSAR figures 3.6.1-1 through 3.6.1-6 describe the details of key parameters for the event including CR position, neutron flux, reactor power, reactivity, HPB pressure and fuel temperatures.

Loss of Primary Flow DBA

PSAR section 3.6.2 describes the LOPF DBA analysis and the events composing PRA sequence mapped from MTLPF DBE with only SR systems credited and rods failing to insert. The IE for this DBA is loss of primary helium flow due to simultaneous trip of both circulators at full power. The possible causes for the IE include spurious trips via signals from DCS, IPS, or

RCS; common cause failures of the circulators or their support systems; or an operator-induced trip. Following the IE, the control and shutdown rods fail to insert, and SSS is assumed to fail. The reactor shuts down on inherent reactivity feedback with conduction cooldown to the reactor cavity and SST passive heat tanks. The event ends with the reactor shutdown to decay power level with intact HPB and long-term cooling (PLOFC) provided by passive RCCS. The PSAR notes that at the end of the short-term event analyzed in Flownex, the fuel temperatures and helium pressures are still increasing. The final temperatures and pressures; however, are bounded by the long-term cooldown PLOFC condition described in PSAR section 3.6.10.

The MTLPF DBA results in a total radiation dose (TEDE) at the EAB of less than 1 rem, below the 25-rem limit for DBAs. PSAR table 3.6.2-1 provides details of the radionuclide release for the event, including the long-term PLOFC event. The transient portion of the MTLPF DBA results in a slight increase in the radiation dose at the EAB due to fuel heat up, and the primary dose contributor is the normal HPB leakage from circulating radioactive inventory during the long-term PLOFC. PSAR figures 3.6.2-1 and 3.6.2-2 describe the details of key parameters for the event including fuel temperatures, and HPB pressure.

Small HPB Depressurization DBA

PSAR section 3.6.3 describes the SD DBA analysis and the events composing the PRA sequence mapped from the SD BDBE with only SR systems credited and rods failing to insert. The acceptability of mapping the SD BDBE to the SD DBA is discussed earlier in this section.

The IE for this DBA is a varying break or leak of up to 10 mm in the HPB at full power. The 10 mm break size corresponds to the small instrument line on the HPB. Following the IE and failure to isolate the leak, the control and shutdown rods fail to insert. The event leads to slow depressurization of the primary system with release of radionuclides due to fuel heat up. The HPB depressurizes as a result of the break, leading to the RPS low pressure trip setpoint being reached. Main loop cooling and SSS are assumed to fail. The event assumes a helium pumpdown failure where the transfer of helium to the storage tank fails. The reactor shuts down on inherent reactivity feedback with no RB heating, ventilation, and air conditioning (HVAC) filtration available. The event ends with the reactor shutdown with DLOFC provided by passive RCCS. PSAR section 3.6.11 describes the long-term DLOFC state for the event.

The NRC staff reviewed the PSAR description of the event, which includes a description of a break size and location sensitivity study. The study shows that a HPB pressure break in the defueling chute differs from other locations due to the differential pressure being smaller for a break at that location. The reactor power and offsite dose for all break locations are identical. The sensitivity study also shows that a 5 mm break size represents a more conservative case for maximum fuel temperatures as compared to the 3 mm and 10 mm break. The 3 mm break leads to the highest offsite dose due to the longer depressurization period at elevated temperatures. PSAR table 3.6.3-1 provides the details of the sensitivity study performed for different break sizes, locations and systems active/inactive for the event. The table also shows that RCCS is not available for long-term cooling for SD DBA case 5, with a 10 mm break at the reactor inlet.

PSAR table 3.2-5 shows a total radiation dose (TEDE) at the EAB for the SD DBA below the 25-rem (0.25 Sv) limit for DBAs. PSAR table 3.6.3-2 provides details of the radionuclide release for the SD DBA case 4, which resulted in the highest offsite dose. PSAR figures 3.6.3-1 through

3.6.3-6 describe the details of key parameters for each of the sensitivity cases analyzed for the event including the mass flow rate, inlet pressure, maximum and average fuel temperatures, reactivity, and power. PSAR figure 3.6.3-7 provides a comparison of the DLOFC extreme case maximum fuel temperature to the SD DBA maximum fuel temperatures to show that DLOFC temperatures remain bounding.

The NRC staff notes that PSAR figure 3.6.3-3 shows maximum fuel temperatures for the SD DBA that go above 1600°C. Based on the NRC staff's literature review (PNNL-31427), higher dose consequences are expected beyond 1600°C. The NRC staff confirmed in the regulatory audit that the dose consequence models account for higher releases at these elevated temperatures, thus providing reasonable assurance that preliminary the event consequences for SD DBA are sufficient to assure safety margins during normal operations and transient conditions for the preliminary LMGS design. At the OL application stage, the NRC staff will confirm the accuracy of the consequence models based on the planned testing to demonstrate performance of the Xe-100 fuel at these elevated temperatures as discussed in LMGS R&D program (PSAR chapter 13).

Medium HPB Depressurization DBA

PSAR section 3.6.3 describes the MD DBA analysis and the events composing the PRA sequence mapped from MD DBE with only SR systems credited and rods failing to insert.

The IE for this DBA is a variable break size between 25 mm and 65 mm in diameter, where the 65 mm break size corresponds to the largest pipe in the HPB and 25 mm corresponds to the approximate geometric mean of the 65 mm and 10 mm break used in the MD DBE. Following the IE and failure to isolate the leak, the control and shutdown rods fail to insert. The event leads to rapid depressurization of the primary system with release of radionuclides due to fuel heat up. The main loop cooling and SSS are assumed to fail. The event assumes a helium pumpdown failure where the transfer of helium to the storage tank fails. The reactor shuts down on inherent reactivity feedback with no RB HVAC filtration available. The event ends with the reactor shutdown to decay power level with long-term cooling (DLOFC) provided by passive RCCS. PSAR section 3.6.11 describes the long-term DLOFC state for the event.

The NRC staff reviewed the PSAR description of the event, which includes a description of a break size and location sensitivity study. PSAR table 3.6.4-1 presents a sensitivity study for two break sizes (65 mm and 25 mm) performed at six locations: reactor inlet, defuel chute, SG outlet, control rod drive mechanism stand tube, FHS inlet, and reactor vessel lid. The table provides key parameters such as fuel temperatures, reactivity, RPV temperature, and mass flow rate for each sensitivity case. Based on the results presented in PSAR table 3.6.4-1, the NRC staff notes that the break size has a limited impact on offsite dose for the events analyzed and the break in the defueling chute differs in transient response from other locations due to the differential pressure being smaller for a break at that location.

PSAR table 3.2-5 shows a total radiation dose (TEDE) at the EAB for the MD DBA below the 25-rem (0.25 Sv) limit for DBAs. PSAR table 3.6.4-2 provides details of the radionuclide release for the event. PSAR figures 3.6.4-1 through 3.6.4-7 describe the details of key parameters for each of the sensitivity cases analyzed for the event including inlet pressure, maximum and average fuel temperatures, reactivity and power. The NRC staff notes that the maximum fuel temperature for the 25 mm break at the reactor inlet presented in PSAR figure 3.6.4-7 is still rising at the end of the short-term event. The long-term cooling for this event behaves similar to

the long-term cooling for the SD DBA, and PSAR figure 3.6.3-7 provides a comparison of the DLOFC extreme case to the SD DBA which shows that DLOFC extreme fuel temperatures will remain bounding.

Loss of Steam Generator Feedwater DBA

PSAR section 3.6.5 describes the LOFW DBA analysis and the events composing the PRA sequence mapped from MTLFW BDBE with only SR systems credited and rods failing to insert. The acceptability of mapping the MTLFW BDBE to the MTLFW DBA is discussed earlier in this section.

The IE for this DBA is a total loss of main feedwater to both SG bundles due to tripping of the HPFWP motors in one reactor module. The loss of feedwater flow results in loss of heat transfer capability from primary to secondary side. The loss of feedwater flow leads to an RPS trip, but the control and shutdown rods fail to insert. Subsequently, the circulators trip on high He/FW flow ratio. The HPB pressure rises initially but the relief set point is not reached for the event. The analysis assumes failure of the feedwater pumps and the SSS system. Passive RCCS provides long-term cooling for the event. The reactor shuts down on inherent reactivity feedback. The event ends in a safe, stable shutdown with passive cooling in PLOFC mode provided via the RCCS. PSAR section 3.6.10 describes the PLOFC event.

PSAR figure 3.6.5.2 shows that the helium circulator speed does not go below 500 rpm even after the trip, which is a result of not including a circulator friction model in the Flownex code. The helium flow ceases when the circulators trip, as shown in PSAR figure 3.6.5-1. PSAR figure 3.6.5-1 shows flow increasing at approximately 3 minutes into the transient due to residual feedwater flashing into steam as pressure decreases to its boiling point.

The MTLFW DBA results in a total radiation dose (TEDE) at the EAB below the 25-rem (0.25 Sv) limit for DBAs. PSAR table 3.6.5-1 provides details of the radionuclide release for the event, including the long-term PLOFC event. The transient portion of the MTLFW DBA does not result in a radiation dose at the EAB, and the primary dose contributor is the normal HPB leakage from circulating radioactive inventory during the long-term PLOFC. PSAR figures 3.6.5-1 through 3.6.5-6 describe the details of key parameters for the event including feedwater mass flow, circulator speed, HPB pressure, power, reactivity, and core temperatures. The maximum fuel temperature for the DBA presented in PSAR figure 3.6.5-6 is still rising at the end of the short-term event, but overall the fuel temperature is bounded by the long-term PLOFC event presented in PSAR section 3.6.10.

Main Feedwater Line Break DBA

PSAR section 3.6.6 describes the MFLB DBA analysis and the events composing the PRA sequence mapped from the MFLB DBE with only SR systems credited and rods failing to insert.

The IE for the MFLB DBA is a double guillotine break of the combined feed line header. An RT, initiated by the RPS on high cold helium temperature at the SG outlet, fails to insert control rods and shutdown rods. Following the failure of rod insertion, the helium circulators trip and SSS is assumed to fail. The reactor shuts down on inherent reactivity feedback. At the end of the event, the reactor shuts down to decay power levels with an intact HPB, and passive RCCS provides long-term cooling. PSAR section 3.6.11 describes the long-term PLOFC state for the event.

The MFLB DBA results in a total radiation dose (TEDE) at the EAB below the 25-rem (0.25 Sv) limit for DBAs. PSAR table 3.6.6-1 provides details of the radionuclide release for the event, including the long-term PLOFC event. The transient portion of the MFLB DBA does not result in a radiation dose at the EAB, and the primary dose contributor is the normal HPB leakage from circulating radioactive inventory during the long-term PLOFC. PSAR figures 3.6.6-1 through 3.6.6-6 describe the details of key parameters for the event including reactivity, power, HPB pressure, circulator speed, check valve position, and core temperatures. The maximum fuel temperature for the DBA presented in PSAR figure 3.6.6-6 is still rising at the end of the short-term event, but the fuel temperature is bounded by the long-term PLOFC event presented in PSAR section 3.6.10.

Main Steam Line Break DBA

PSAR section 3.6.7 describes the MSLB DBA analysis and the events composing the PRA sequence mapped from the MSLB BDBE with only SR systems credited and rods failing to insert. The acceptability of mapping the MSLB BDBE to the MSLB DBA is discussed earlier in this section.

The IE for the MSLB DBA is a guillotine break downstream of the tee joining the supply lines from two separate SG tubes. A break at this location maximizes the rupture size and affects both tube bundles. The SG generator temperature and pressure drop suddenly, and the reactor responds to an increased steam demand by settling at a new power level above 100 percent, but below the RPS setpoint. This condition persists as long as the feedwater is available. This operation exhausts the feedwater supply in the deaerator tanks and the RPS generates a trip on loss of feed supply. The rods fail to insert and the trip is unsuccessful. The feedwater pumps trip with feed supply running out and this causes a sudden steam temperature spike before stabilizing. The reactor shuts down on inherent reactivity feedback and relief setpoint is not reached. At the end of the event, the reactor shuts down to decay power levels with an intact HPB, and passive RCCS provides long-term cooling. PSAR section 3.6.10 describes the long-term PLOFC state for the event.

The MSLB DBA results in a total radiation dose (TEDE) at the EAB below the 25-rem (0.25 Sv) limit for DBAs. PSAR table 3.6.7-1 provides details of the radionuclide release for the event, including the long-term PLOFC event. The transient portion of the MSLB DBA does not result in a radiation dose at the EAB, and the primary dose contributor is the normal HPB leakage from circulating radioactive inventory during the long-term PLOFC. PSAR figures 3.6.7-1 through 3.6.7-6 describe the details of key parameters for the event including mass flow, main steam temperature, steam pressure, deaerator tank inventory, reactivity, power, maximum core temperatures and HPB pressure. The maximum fuel temperature for the DBA presented in PSAR figure 3.6.6-6 is still rising at the end of the short-term event, but is bounded by the long-term PLOFC event presented in PSAR section 3.6.10.

Steam Generator Tube Leak DBA

PSAR section 3.6.8 describes the SGTL DBA analysis. As discussed in PSAR section 3.4, since the tube rupture event is more limiting than the similar tube leak event, a description of the SSGTL DBE sequences is not provided and is not directly mapped to a DBA.

The NRC staff reviewed PSAR section 3.6.8, which states that the SSGTL DBA and SSGTR DBA have identical ESs, except that the SSGTR DBA is a more conservative double-ended

16 mm guillotine break modeled as two breaks of 16 mm as compared to the 10 mm break for the SSGTL DBA. Since the SSGTL-DBA and SSGTR-DBA events have nearly identical progression with the results for SSGTR DBA results bounding the SSGTL DBA, the PSAR only presents detailed results for the SSGTR DBA. Oxidation rates are also higher for the SSGTR DBA due to the higher rate of water ingress.

The NRC staff finds this mapping acceptable since both events have similar progression with the larger break size for the SSGTR DBA leading to bounding or similar fuel temperatures and bounding or similar dose consequences.

Steam Generator Tube Rupture DBA

PRA section 3.6.9 describes the SGTR DBA analysis and the events composing the PRA sequence mapped from the SSGTR DBE with only SR systems credited and rods failing to insert. The PSAR states that the evaluation of SSGTR uses worst-case values for the PKPIRT uncertainty parameters.

The IE for this DBA is a double-ended guillotine break and is modeled as two breaks of 16 mm in one SG tube near the top of a SG bundle. Following initiation, the IPS and RPS trip the reactor on HPB moisture, however the control and shutdown rods fail to insert. Subsequently, the circulators trip and SG isolation sequence begins. The isolation of the affected SG has two steps; feedwater-side isolation and steam-side isolation. The turbine bypass valve is assumed to fail in the closed position. The PSAR states that although the turbine bypass signal is provided by the NSRST IPS, the turbine bypass signal is included in the DBA analysis to maximize the HPB pressure rise. The NRC staff finds that including the turbine bypass signal in the ES is acceptable as crediting the NSRST IPS leads to bounding HPB pressure. Post the SG isolation, the water in one of the SG train bundles leaks into the HPB until the pressure equalizes. The reactor shuts down on inherent reactivity feedback. The HPB relief setpoint is reached and the HPB relief valve opens and closes to relieve the pressure. Since additional relief lifts are expected during long-term cooldown of the event, the valve is assumed to fail open. On the secondary side, the safety relief valves on the intact SG lines lift multiple times to relieve pressure in isolated steam piping. At the end of the event, the reactor shuts down to decay power levels with HPB reaching atmospheric levels, and passive RCCS provides long-term cooling. PSAR section 3.6.11 describes the long-term DLOFC state for the event.

PSAR figure 3.6.9-13, which presents the core reactivity as function of time during the SSGTR DBA, shows that the control rods and steam add positive reactivity to the event. LME clarified that this display is due to a limitation of the Flownex code. The PSAR states that positive reactivity added is from steam contribution only since the rods are not credited (insertion or withdrawal) for the DBA.

The SSGTR DBA results in a total radiation dose (TEDE) at the EAB below the 25-rem (0.25 Sv) limit for DBAs. PSAR table 3.6.9-1 provides details of the radionuclide release for the event, including the long-term DLOFC event. PSAR figures 3.6.9-1 through 3.6.9-12 describe the details of key parameters for the event including humidity, rod depth, mass flow rate, valve positions, main steam mass flow, tube rupture mass flow, power, reactivity, core temperatures and HPB and SG outlet pressures. The maximum fuel temperature for the DBA presented in PSAR figure 3.6.9-9 is still rising at the end of the short-term event but is bounded by the long-term DLOFC event presented in PSAR section 3.6.11, making it conservative to use the DLOFC results for the long-term contribution rather than modeling the specific event.

Pressurized Loss of Forced Cooling DBA

PSAR section 3.6.10 describes the PLOFC analysis for DBA events and the long-term phase of several DBAs combined with the short-term transient phase to determine the dose consequence for the given event. The PLOFC DBA assumes the HPB remains intact but there is no active cooling available and core cooling is provided primarily through passive RCCS system and buoyancy-driven helium convection.

The NRC staff reviewed the description of the PLOFC DBA, which states that the event does not credit rod insertion as a conservative measure to produce higher temperatures and bounding dose consequences. The NRC staff finds this acceptable because the assumption leads to bounding dose consequences and there is no concern for recriticality since the actual designed system response would involve rod insertion.

The PLOFC DBA event is modeled using the GOTHIC model for the Xe-100 reactor and is simulated for 30 days as the reactor transitions to a stable long-term plant response as described in PSAR section 3.1.3. The total dose consequence for the event is the sum of the doses from the short-term event modeled in Flownex and long-term plant response modeled in GOTHIC, with both simulations starting the IE at time zero (i.e. the short-term plant response from Flownex is not an input to GOTHIC for long-term response). This modeling approach produces an overall conservative dose estimate without modeling each DBA through its long-term evolution.

The PLOFC DBA analysis uses the PKPIRT extreme value case with the most conservative value for the parameters presented in PSAR table 3.6.10-1. These parameters were determined to have the highest impact in the PKPIRT.

The NRC staff notes that the PKPIRT parameters used in the PLOFC and DLOFC cases are described in the TSAM TR (ML25206A049), which is IBR. In the SE of the TSAM TR, the NRC staff were unable to determine that the PIRT panel was informed by sufficient or suitable information for PKPIRT. Further, the NRC staff were unable to determine in the TSAM SE that the uncertainty analysis methodology applied to the PKPIRT was appropriate since the acceptance criteria described in the TR appeared to restrict the number of phenomena, and the parameters were considered without providing appropriate justification. As such, conditions 5 and 14 applied to the TSAM TR state the need for identification of the remaining scope of work as a part of a relevant R&D program consistent with 10 CFR 50.34(a)(8). However, as summarized in PSAR table 1.4-11 and based on the criteria presented in PSAR chapter 13, the applicant states that these conditions do not meet the definition of the R&D program under 10 CFR 50.34(a)(8). The NRC staff's review of PSAR table 1.4-11 found that the applicant identifies the outstanding scope of work associated with all relevant TSAM conditions - including gaps in PIRT and commits to completing these activities at the OL application stage. The technical or design information associated with all relevant TSAM conditions, as may be required to complete the safety analysis, will be supplied in the FSAR at the OL application stage.

The PLOFC DBA dose combined with doses from the short-term transient phase of respective DBAs, as summarized in PSAR table 3.2-5, results in a total radiation dose (TEDE) at the EAB below the 25-rem (0.25 mSv) limit. PSAR figures 3.6.10-1 through 3.6.10-6 describe the details of key parameters for the event including HPB pressure, maximum and average fuel temperature, and reactor power.

PSAR figure 3.6.10-4 shows that, for the PLOFC DBA, a maximum fuel temperature of 1760°C is reached for the bounding case (CRW event) at approximately 1.3 hours. Based on the NRC staff's literature review, higher dose consequences are expected beyond 1600°C. The NRC staff confirmed in the regulatory audit that the consequence models account for higher releases at these elevated temperatures, thus providing reasonable assurance that preliminary event consequences for the PLOFC DBA are sufficient to assure safety margins during normal operations and transient conditions for the preliminary LMGS design.

At the OL application stage, the NRC staff will confirm the accuracy of the consequence models based on the planned testing to demonstrate the performance of the Xe-100 fuel at these elevated temperatures as discussed in the LMGS R&D program (PSAR chapter 13).

Depressurized Loss of Forced Cooling DBA

PSAR section 3.6.11 describes the DLOFC analysis for DBA events and the long-term phase of several DBAs combined with the short-term transient phase to determine the dose consequence for the given event. The DLOFC DBA assumes that the HPB does not remain intact due to leaks or breaks, the reactor is depressurized to atmospheric levels, no active cooling is available and passive RCCS provides long-term core cooling.

The NRC staff reviewed the description of the DLOFC DBA that states that it does not credit rod insertion as a conservative measure to produce higher temperatures and bounding dose consequences. The NRC staff finds this acceptable since the assumption leads to bounding dose consequences and there is no concern for recriticality since the actual designed system response would involve rod insertion.

The DLOFC DBA event is modeled using the GOTHIC model for the Xe-100 reactor and is simulated for 30 days as the reactor transitions to a stable long-term plant response as described in PSAR section 3.1.3. The total dose consequence for the event is the sum of the doses from short-term events modeled in Flownex and long-term plant responses modeled in GOTHIC, with both simulations starting the IE at time zero (i.e. the short-term plant response from Flownex is not an input to GOTHIC for long-term response). This modeling approach produces an overall conservative dose estimate without modeling each DBA through its long-term evolution.

The DLOFC DBA analysis uses the PKPIRT extreme value case with the most conservative value for the parameters presented in PSAR table 3.6.11-1. The NRC staff's evaluation and acceptability of the use of PKPIRT parameters is in the PLOFC DBA section of this SE. PSAR section 3.4.8 provides the basis for the selection of 3 mm and 65 mm break sizes for the DLOFC evaluation. The DLOFC DBA evaluation uses the 65 mm break in the refueling tube as the IE.

The DLOFC DBA dose combined with doses from the short-term transient phase of respective DBAs, as summarized in PSAR table 3.2-5, results in a total radiation dose (TEDE) at the EAB below the 25-rem (0.25 Sv) limit. PSAR figures 3.6.11-1 through 3.6.11-6 describe the details of key parameters for the event including HPB pressure, maximum and average fuel temperature, and reactor power.

PSAR table 3.6.11-2 shows results for peak fuel temperatures for the nominal case (1803°C) and the extreme case (1963°C), along with temperature increase from applying uncertainties

associated with each of the PKPIRT parameters (fuel effective conductivity, power, decay heat, and reflector emissivity). The PSAR concludes that comparing the extreme case temperature increase to the sum of the individual cases demonstrates that local linearity of the fuel temperature in response to PKPIRT parameters applied and the absence of significant interaction effects within defined uncertainty ranges allows the uncertainties to be treated as independent. The PSAR then applies 'square root of sum of the squares' to combine uncertainties and lists the peak fuel temperature with uncertainty as 1885°C.

The NRC staff notes that local linearity of the fuel temperatures does not imply that the statistical independence of uncertainties can be assumed. While it is reasonable to assume that there will be no significant interaction effects for the parameters defined, the analysis to show that the uncertainties to be treated as independent is not available at the CP stage. As discussed in the PLOFC DBA section of this SE, the applicant has committed to performing remaining work to identify and remedy gaps in the PIRT and commits to completing these activities at the OL application stage.

The NRC staff notes that PSAR figure 3.6.11-2 shows that the maximum fuel temperatures for the DLOFC DBA exceed 1800°C. Based on the NRC staff's literature review (PNNL-31427), higher dose consequences are expected beyond 1600°C, and 1800°C is generally characterized as the severe accident temperature. The NRC staff confirmed in the regulatory audit that the consequence models account for higher releases at these elevated temperatures, thus providing reasonable assurance that the event consequences for DLOFC DBA are sufficient to assure safety margins during normal operations and transient conditions for the preliminary LMGS design. At the OL application stage, the NRC staff will confirm the accuracy of the consequence models based on the planned testing to demonstrate performance of the Xe-100 fuel at these elevated temperatures, as discussed in the LMGS R&D program (PSAR chapter 13).

Seismic DBA

PSAR section 3.6.12 describes the seismic analysis for DBA events and the preliminary scoping analysis to evaluate effects of a hypothetical seismic event at LMGS.

The seismic analysis for DBA events assumes a seismic event that results in a LOOP which trips the turbine, helium circulators, and the feedwater pumps in all four modules. Positive reactivity insertion due to compacting of the pebble bed is modeled, increasing the reactor power above the trip set point. Consistent with other DBAs, the event assumes failure of control and shutdown rods to insert.

The NRC staff reviewed PSAR table 3.6.12-2, which presents the analysis of three scenarios for the seismic DBA: a 10 mm diameter HPB leak in the reactor inlet piping (seismic-A), a 95 mm diameter break at the top of the RPV (seismic-B), and a no break case (seismic-C). The overall dose consequence for the seismic DBA is determined based on the worst combination for a four-module station. The table also presents a description of the three seismic events along with the status of specific SSCs for seismic event modeling. The PSAR states that the 95 mm break is based on the double-ended guillotine break of the largest pipe in the HPB (67 mm) and clarifies that the slight difference in the diameter for the largest pipe assumed here versus in the MD DBE (67 mm vs. 65 mm) are expected to yield similar dose results. The NRC staff agrees that the minor difference in diameter is not expected to change the offsite dose based on the analysis presented for different break sizes in MD LBE events.

The PSAR states that during the event, the pebble bed compacts from its initial packing fraction of 0.61 to a pack fraction of 0.64, leading to a step increase in reactivity of 2.46 mk (246 pcm). This value is based on the experiments performed at Idaho National Laboratory (Report INL/EXT-11-2307). The PSAR states that the impact of compaction on thermodynamic properties is not modeled because the analysis of M-tech SANA experiments demonstrates a minor impact of compaction on thermodynamic properties. The NRC staff finds this acceptable based on the experimental basis provided in the referenced INL report.

The seismic DBA analysis assumes that all seismically qualified LMGS systems do not fail during the event, non-safety-related and NSRST SSCs are not available, and the rods are not credited. The RCCS remains intact to support passive heat removal. For cases with an HPB break, the short-term portion of the calculation ends when the HPB pressure is equal to the RB pressure. For the case with an intact HPB, the short-term calculation ends with the primary circulation flow reduced to the point where the helium check valves have fully closed and heat is no longer being removed via the SG. The long-term state of the plant is then analyzed as the PLOFC DBA or DLOFC DBA case depending on whether the HPB is intact.

The seismic DBA total radiation dose (TEDE) at the EAB is determined by multiplying the worst single module case (a 95 mm break) by a factor of four. This is conservative since the EAB dose from a seismic event affecting all four modules would generally result in a dose less than four times that of single module due to the impact of the dispersion factor.

The PSAR states that the TEDE at the EAB for the limiting seismic-B DBA is less than the 25-rem (0.25 Sv) limit. PSAR table 3.6.12-3 provides details of the radionuclide release for the limiting seismic-B event, including contribution from the long-term DLOFC. PSAR figures 3.6.12-1 through 3.6.12-12 describe the key plant parameters for the limiting event which show that fuel temperatures remain bounded by the long-term DLOFC.

PSAR table 3.2-4 shows that several events that are not explicitly modeled in the PSAR are mapped to the seismic-B DBA (95 mm break) event. The NRC staff concluded in section 3.2 of this SE that it is acceptable to map certain events to the seismic-B DBA event based on the bounding dose consequence and the applicant's commitment to perform a fully refined LBE evaluation and a final ES family grouping in support of a future OL application.

Impacts of Combustible Gases during DBAs

The generation of combustible gases during DBAs for the Xe-100 reactor occurs primarily due to steam or water ingress events causing graphite-steam reaction or from air ingress during system depressurization events.

PSAR section 3.6.9.2 states that for combustible gas production, the SSGTR DBA is the most limiting for all tube rupture or tube leak events. The graphite-steam interaction due to the event can lead to the production of flammable gases including hydrogen and carbon monoxide. The PSAR states that an analysis was performed to demonstrate that the resulting chemical reactions from this event do not produce a flammable mixture.

Similarly, for the system depressurization events, PSAR section 3.6.11.5 states that the air taken in during the long-term cooling event can react with graphite to produce flammable gases. The PSAR states that an analysis performed using GOTHIC code demonstrated that over a 30-

day long-term cooling period event, if all the oxygen intake is assumed to be consumed by the graphite resulting in carbon monoxide production, the gaseous mixture in the HPB will remain below the lower flammability limit of the carbon monoxide.

During the regulatory audit, the NRC staff reviewed the SGTR oxidation analysis (006367, 'Xe-100 SGTR Oxidation Analysis Flammability Limit Report') and the DLOFC air ingress analysis (010506, 'Xe-100 DLOFC Air Ingress Analysis Service Receipt Inspection Report') to better understand the analyses performed.

Accurate characterization of air ingress during a DLOFC event is essential for determining whether the current design provides adequate mitigation of consequences for air ingress. More refined analyses may reveal that additional design strategies are necessary to further limit the impact of air ingress. Existing one-dimensional modeling approaches in GOTHIC or Flownex have not been validated for predicting multi-dimensional flow behavior, such as mixing patterns and specific air ingress mechanisms at the break (e.g., density driven flow, natural circulation, flow oscillations, or stratified flow), that are identified in literature on this topic. The PSAR designates a 65 mm break at the refueling tube in the control rod drive mechanism region as the bounding case for fuel heat up, based on the assumption that larger breaks would not significantly alter depressurization rates or fuel heat-up behavior. However, larger breaks or breaks occurring at other locations, such as the bottom of the reactor vessel, could lead to significantly higher air ingress and more extensive oxidation of the bottom graphite support structures. A more detailed analysis of potential break sizes, locations, and associated flow phenomena are required to appropriately evaluate whether additional design or mitigation strategies are needed to address air ingress effects during a DLOFC event. Similarly, the GOTHIC and Flownex modeling approach for steam ingress related phenomena was not adequately discussed in the GFQ TR (e.g., identification of code models and correlations and validations).

The PSAR commits to evaluating break size, location, orientation and multi-dimensional flow effects on air ingress and graphite oxidation at the OL application stage. The NRC staff finds this identification of remaining scope of work and a commitment to complete it at the OL application stage to be acceptable in section 3.1.3 of this SE.

Therefore, in accordance with 10 CFR 50.34(g), which requires an applicant for a CP to include the analyses and the descriptions of the equipment and systems required by 10 CFR 50.44(d) as a part of an application, the NRC staff determined that the information provided in the PSAR for the CP application is acceptable.

Evaluation of the Exceptions to NEI Guidance for DBAs

The PSAR identifies three exceptions to the guidance in NEI 18-04, which states that deterministic DBA evaluations should be derived from the corresponding DBEs. The exceptions are discussed below.

- For the SD event, the PSAR states that the PRA analysis identified three DBEs, but no specific safety analysis is presented for these DBEs since the end state indicates that the offsite dose for the DBE would be the same as the offsite dose for the BDBE that was explicitly evaluated.

During the regulatory audit, the NRC staff requested additional details pertaining to this determination since PSAR table 3.2-2 shows that the DBEs (SD-076, SD 084, and SD 098) involve unsuccessful plant rebalance, requiring modules 2 - 4 to trip, as opposed to the SD-074 DBE, which has successful plant rebalancing. The applicant provided justification that the selection of event mapping was based on the bounding offsite dose for the LBEs since all the LBEs in question experience the same size HPB break. Since the long-term cooling for the SD-74 event is provided by passive RCCS, as opposed to SSS for the DBEs, this event leads to a higher offsite dose and aligns better with the SD DBA which also ends in passive RCCS long-term cooling. The NRC staff finds this acceptable based on the ES for the SD-74 BDBE being closer to the SD DBA and offsite dose for the event bounding other similar DBEs.

- For the LOFW event, the PSAR states that the MTLFW BDBE, analyzed in PSAR section 3.5.5, was used as the basis for the DBA instead of the similar DBE identified by the PRA in PSAR table 3.2-2. This is due to the end state for the events indicating that the offsite dose for the DBE would be the same as the offsite dose for a BDBE.

- The NRC staff notes that the dose consequence for the MTLFW-02 DBE and MTLFW-03 BDBE identified in PSAR table 3.2-2 are the same. The only difference in event progression is that MTLFW-02 ends in active RCCS while MTLFW-03 ends in passive RCCS, consistent with the DBA. Because active RCCS is NSRST, the DBA for MTLFW-02 would have an identical ES to MTLFW-03. PSAR table 3.2-4 appropriately identifies both events as being mapped to MTLFW DBA. Since detailed evaluation of the MTLFW-02 DBE is not provided in the PSAR, the NRC staff finds that mapping of the MTLFW-03 BDBE to the MTLFW DBA is acceptable because both events have the same dose consequence and identical event progression when considering only SR SSCs.

  The PSAR states that the PRA analysis did not identify any MSLB DBEs. PSAR table 3.2-2 identifies two AOOs (MSLB-01, MSLB-07) and two BDBEs (MSLB-08, MSLB-12) for the MSLB IE. The PSAR states that since the upper bound frequency for the MSLB-08 BDBE was approximately equal to the lower end of the DBE range, the BDBE was used for the DBA. MSLB-01 and MSLB-07 both have 5th percentile frequencies in the DBE region and would also be mapped to a DBA, according to NEI 18-04 guidance. In PSAR table 3.2-4, LME does map the MSLB-01 ES to the MSLB DBA due to its 5th percentile frequency (6.30E-05/year) but does not include MSLB-07 (5th percentile frequency of 3.66E-05/yr). The NRC staff reviewed this exception and the impact of MSLB-07 not being mapped to the DBA and finds them acceptable for the CP stage in that all three events have the same ES with the exception of the final cooling state, with each relying on various forms of non-safety-related cooling, which would default to passive RCCS cooling in the DBA analysis, consistent with the MSLB DBA.

Further, PSAR section 3.2 discusses LBEs which are not explicitly evaluated in PSAR sections 3.3, 3.4 and 3.5. These events include DBEs and other LBEs, which did not get mapped into corresponding DBAs. In PSAR section 3.2, the applicant committed to performing a fully refined LBE evaluation and a final ES family grouping in support of a future OL application. The NRC staff finds this to be acceptable for the preliminary design at the CP stage in SE section 3.2.

Confirmatory DBA Evaluations using MELCOR

Section 3.1.4 of this SE provides details of the confirmatory analysis performed by the NRC staff and its contractors (Sandia National Laboratory) using the SCALE simulation code and the MELCOR accident analysis code to independently assess the transient behavior and radionuclide release estimates.

The MELCOR analyses performed as part of the confirmatory analysis focuses on temperature distributions for accident scenarios, particularly temperature distributions in TRISO, fuel pebbles and core components for the limiting DLOFC event. The MELCOR analyses analyzed the impact of variation in the PKPIRT parameters from the PSAR for different cases (e.g. power, decay power, thermal conductivities, emissivity). The analyses also evaluated the influence of expected variations in other input parameters (e.g., helium flow, inlet temperature, main steam pressure, feedwater mass flow, feedwater temperature, and reactivity coefficients).

The MELCOR model for the Xe-100 reactor was developed using the geometric details from the Flownex model provided by the applicant during the regulatory audit. The MELCOR model was developed with a very detailed control volume, comparable to the Flownex model used for the CP application, consisting of twenty-seven axial levels, thirty radial rings, thirteen different material compositions including five different graphite types, and multiple neutron leakage pathways. The axial and radial profiles as well as reactivity feedback profiles matched the VSOP profiles from the core design methodology and analysis TR (IBR, ML25071A403). The thermo-physical properties for the reactor, including irradiated and unirradiated graphite conductivities, were taken from the GOTHIC and Flownex TR (IBR, ML25073A097) and MST TR (IBR, ML25073A097), as well as from the reports provided by LME during the regulatory audit. The TRISO heat transfer module included explicit modeling of all the TRISO layers. Nine hot pebble zones were modeled axially at the core centerline to calculate the peak axial fuel temperatures.

Figures 3.6-1 and 3.6-2 of this SE provide a comparison of the MELCOR calculated average and peak fuel temperatures for the DLOFC event with the values presented in PSAR section 3.4.8. The average fuel temperature is useful for characterizing overall fuel performance and estimating temperature-dependent radionuclide release. Histograms of the number of pebbles in each temperature bin were also generated to support the estimation of radionuclide releases. The maximum fuel temperature is important for identifying localized conditions that could challenge the integrity of fission product barriers, which could result in increased release. The DLOFC DBE event was chosen for comparison since it is the limiting long-term event for fuel temperatures.

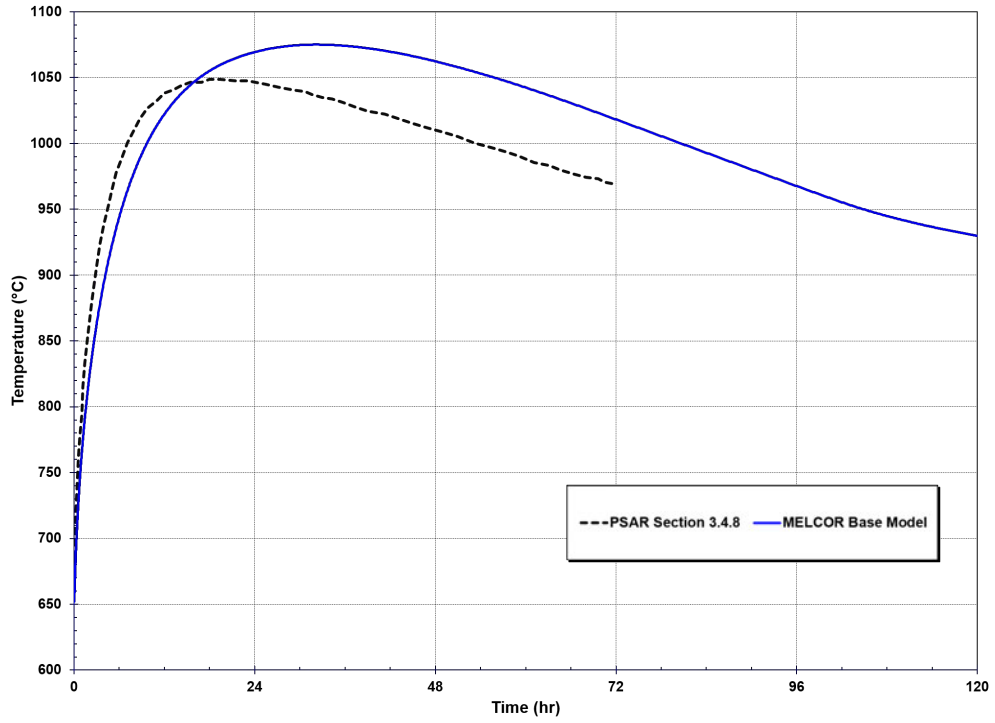

Figure 3.6-1: Comparison of Average Fuel Temperatures

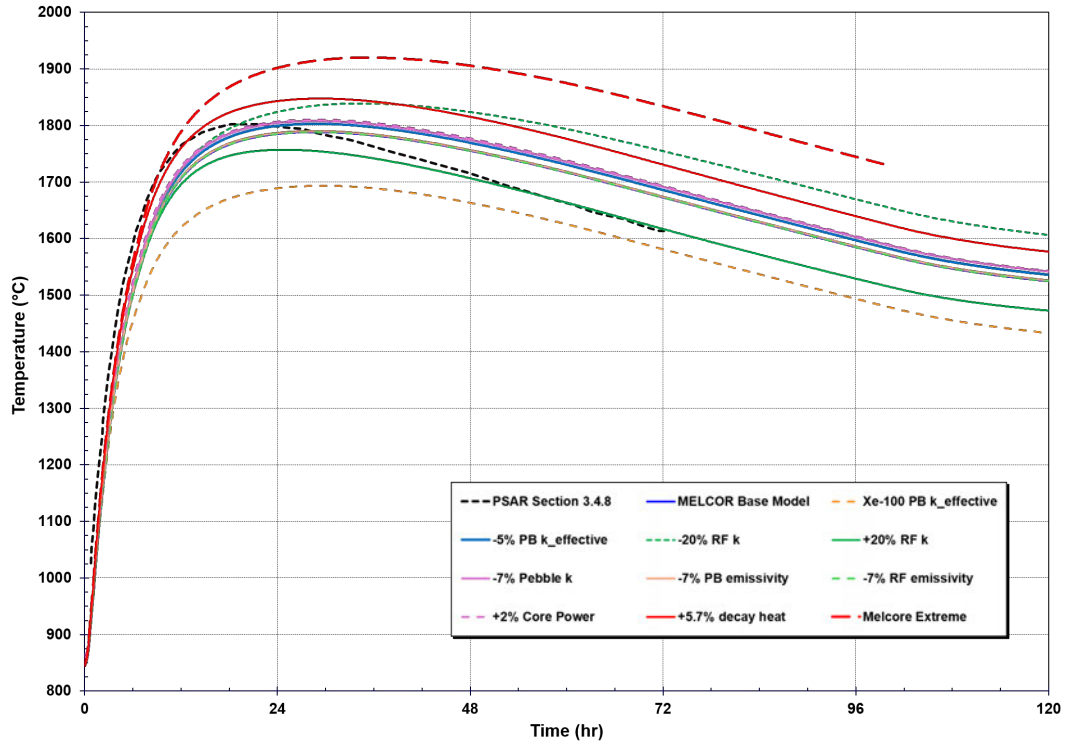

Figure 3.6-2: Comparison of Peak Fuel Temperatures

Figure 3.6-2 of this SE additionally plots MELCOR results for the individual impact of the PKPIRT parameters identified in the LMGS PSAR on the peak fuel temperature. An extreme case is also plotted which combines all of the uncertainties to assess their combined impact on the peak fuel temperature. This case is similar to the 'extreme' case identified in PSAR section 3.6.2.

The results presented show a reasonably good overall agreement between the MELCOR model and the PSAR fuel temperatures. The MELCOR model predicted an overall higher average fuel temperature compared to the PSAR section 3.4.8, as shown in figure 3.6-1 of this SE. The results in figure 3.6-2 of this SE show that varying individual PKPIRT parameters did not have a huge impact on the peak fuel temperatures. The MELCOR base case predicted 1788°C for the peak fuel temperature as compared to the 1803°C prediction in PSAR section 3.4.8. Similarly, the MELCOR extreme case predicted 1920°C for peak fuel temperature as compared to the 1952°C prediction for the extreme case in PSAR section 3.6.11.

The MELCOR analyses also included other scenarios to verify the reasonableness of specific PSAR results (differences in break size for HPB depressurization, CRW, and PLOFC). The results for these analyses aligned well with the results presented in the PSAR.

RPV and core component temperatures are important considerations in the evaluation of transient behavior. The construction code and environmental compatibility assessment for the core components include temperature design limits. Figure 3.6-3 of this SE presents the MELCOR predicted temperatures for the inner and outer RPV wall, the core barrel, and the RB cavity. The results are presented for the 'extreme' DLOFC MELCOR case, which combines all PKPIRT uncertainties to get the bounding temperatures. RPV temperatures are presented for the elevation with the highest wall temperatures, which corresponds to roughly 2m below the top of the pebble bed.

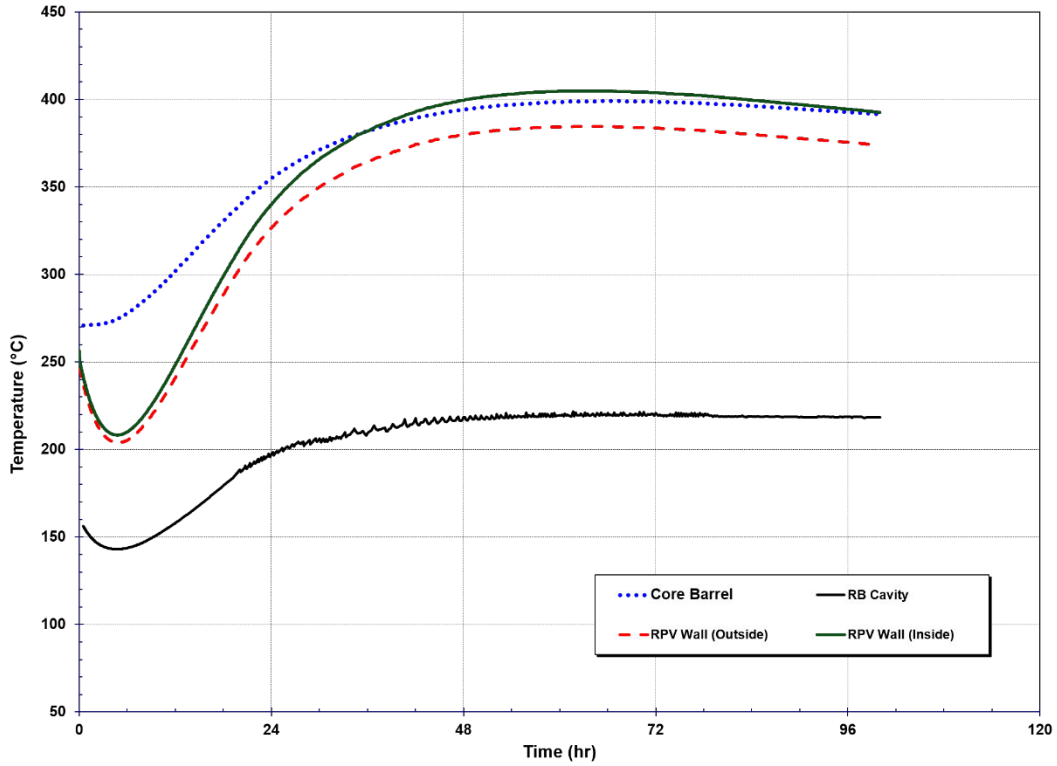

Figure 3.6-3: RPV, Core Barell and RB Cavity Temperatures

PSAR section 6.4.6 lists the design temperature for the RPV as 285°C for normal operations. There is no transient temperature limit provided in the PSAR. PSAR section 6.4.6 lists the material of construction for RPV components as SA-508, and ASME III.5 provides a code temperature limit for SA-508 as 540°C. This temperature is well above the maximum RPV wall temperature predicted for the MELCOR extreme 'DLOFC' case.

The RPV wall and other component temperatures are not provided in the PSAR for the DLOFC cases to allow for a like-for-like comparison. PSAR figure 3.4.1-2 provides RPV temperatures for the bottom, lid, side, and support for the MLOOP DBE. While the MLOOP DBE does not provide a direct comparison to the 'extreme' DLOFC case, it shows that the 'extreme' DLOFC case RPV temperatures bound the DBE event. The peak wall temperature for the MELCOR extreme 'DLOFC' case also bounds the peak RPV wall temperatures presented in PSAR table 3.6.4-1 for the MD DBA.

The results from the confirmatory analyses provide assurance that the models used in the PSAR are reasonable. The NRC staff notes that since the physical models and the thermo-physical properties were adopted from the PSAR and associated calculation files provided during the regulatory audit, the confirmatory analyses performed by the NRC staff assess the implementation of the models and do not provide an independent validation of the full analysis.

3.6.1.1.1 Summary DBA Evaluations and Safety Margins

The NRC staff finds that the evaluation of DBAs in PSAR section 3.6 provides reasonable assurance of safety margins during DBAs because:

1. The applicant used an acceptable EM.

2. The applicant used suitably conservative inputs and boundary conditions for the evaluations.

3. The applicant describes the results of the evaluations including the event initiator, and the narrative of ES is sufficient to understand the plant response and identify the SSCs credited to mitigate the event.

4. The applicant provides the dose consequences, and the details of the radionuclide release are provided where applicable, along with the key parameters of interest to demonstrate margin to the F-C curve.

The NRC staff notes that, for some of the LBEs, no analysis is available in the PSAR and the dose assignment for those events is assigned to the seismic-B DBA. The NRC staff finds this acceptable for the preliminary design at the CP stage because the dose consequence for the events with no analysis are assigned to a higher consequence event, and the applicant committed to evaluate these events at the OL application stage. Similarly, the NRC staff finds the applicant's commitments to perform validation of gaps identified with the PKPIRT and to perform an evaluation of air ingress and graphite oxidation at the OL application stage to be acceptable.

3.6.1.2 Identification of SSCs necessary to prevent accidents and mitigate DBAs

Based on the event descriptions provided in PSAR section 3.6, the NRC staff identified the SSCs credited to prevent and mitigate DBAs. Table 3.6-1 of this SE documents the results of the NRC staff's review of DBAs.

Table 3.6-1: SSCs Credited to Prevent and Mitigate DBAs

PSAR Section	DBE	Initiator	SSCs Credited to Prevent and Mitigate
3.6.1	SCRW DBA	Loss of Offsite Power	NIS, RPS, RCCS
3.6.2	MTLPF DBA	Loss of Primary Flow	NIS, RPS, RCCS
3.6.3	SD DBA	Small HPB Depressurization	NIS, RPS, RCCS
3.6.4	MD DBA	Medium HPB Depressurization	NIS, RPS, RCCS
3.6.5	MTLFW DBA	Loss of SG Feedwater	NIS, RPS, SGS, RCCS
3.6.6	MFLB DBA	Main Feedwater Line Break	NIS, RPS, RCCS

PSAR Section	DBE	Initiator	SSCs Credited to Prevent and Mitigate
3.6.7	MSLB DBA	Main Steamline Break	NIS, RPS, RCCS
3.6.8	SSGTL DBA	Steam Generator Tube Leak	NIS, IPS[1], RPS, SGS, RCCS
3.6.9	SSGTR DBA	Steam Generator Tube Rupture	NIS, IPS[1], RPS, SGS, RCCS
3.6.10	PLOFC DBA	PLOFC Analysis for DBA Events	NIS, RPS, RCCS
3.6.11	DLOFC DBA	DLOFC Analysis for DBA Events	NIS, RPS, RCCS
3.6.12	Seismic DBA	Analysis for Seismic Events	NIS, RPS, RCCS

[1] IPS is an NSRST system but is included in the DBA model to maximize the HPB pressure

Based on the information provided in PSAR section 3.6, the NRC staff finds that sufficient information is provided to identify SSCs credited to prevent and mitigate DBAs because the application identified the SSCs credited to prevent and mitigate DBAs using the information provided in the event descriptions

3.6.1.3 Uncertainty Associated with Performance of SSCs Necessary to Prevent and Mitigate DBA Consequences

RG 1.253, section C.4, “Licensing Basis Events,” describes the connection between evaluating uncertainty associated with performance of SSCs and the identification of safety features and components that require R&D as follows:

> [These criteria] support[s] staff findings associated with 10 CFR 50.35(a)(3), namely, that the application describes the safety features and components that require research and development, and that the applicant will conduct a reasonably designed research and development program to resolve any associated safety questions.

PSAR table 1.4-11 identifies the scope of work needed to address L&Cs associated with the use of the TSAM TR, which includes uncertainty quantifications. Further, an evaluation of the operating band for the input parameters used in the LBE evaluation and the treatment of uncertainties needed to ensure that the LBE evaluations yield conservative results for the 95th percentile consequence is presented in SE section 3.1.1 above. Section 3.6.1 of this SE provides the NRC staff’s evaluation of the acceptability of the uncertainties associated with PKPIRT parameters for the PLOFC and DLOFC analysis.

The applicant provided a commitment to complete the uncertainty quantification at the OL application stage. This provides reasonable assurance that any potential safety questions associated with the adequacy of SSCs and the integral system relied for mitigation of DBAs will be addressed prior to the issuance of an OL.

3.6.2 Conclusion

Based on its review, the NRC staff finds that the DBA information described in PSAR section 3.6 is consistent with the applicable regulatory guidance and meets the regulatory requirements identified in 10 CFR 50.34(a)(1), 10 CFR 50.34(a)(4), 10 CFR 50.34(g), 10 CFR 50.35(a)(2),10 CFR 50.35(a)(3) and 10 CFR 50.44(d). This finding is based on the NRC staff's conclusions that the design will provide sufficient safety margins during normal operations and transient conditions, the applicant has identified the SSCs necessary to prevent and mitigate DBAs, and the applicant has demonstrated an understanding of the uncertainty associated with the performance of SSCs necessary to prevent and mitigate DBA consequences.

3.7 Civil and Structural Analysis Methodologies

PSAR section 3.7 describes the structural analysis and design methods for the LMGS buildings and structures. This PSAR section also covers the seismic classification methodology for SSCs, seismic design basis, design loads and load combinations, seismic analysis of systems and components, and the principal materials for structural analysis and design.

The applicable regulatory requirements for the evaluation of the civil and structural analyses for the buildings and structures are as follows:

- 10 CFR 50.34 (a);
- 10 CFR 50.35; and
- 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants."

The applicable guidance for the evaluation of the civil and structural analyses for the buildings and structures are as follows:

- DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications-Roadmap";
- NUREG-0800, Standard Review Plan (SRP) Chapter 3, "Design of Structures, Components, Equipment, and Systems";
- RG 1.29, Revision 6, "Seismic Design Classification for Nuclear Power Plants," (ML21155A003);
- RG 1.61, Revision 2, "Damping Values for Seismic Design of Nuclear Power Plants," (ML23284A272);
- RG 1.92, Revision 3, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," (ML12220A043);
- RG 1.100, Revision 4, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," (ML19312C677);

- RG 1.122, Revision 1, “Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components,” (ML003739367);

- RG 1.142, Revision 3, “Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments),” (ML20141L613);

- RG 1.199, Revision 1, “Anchoring Components and Structural Supports in Concrete,” (ML19336A079);

- RG 1.243, Revision 0, “Safety-Related Steel Structures and Steel-Plate Composite (SC) Walls for other than Reactor Vessels and Containments,” (ML21089A032); and

- RG 1.233, Revision 0, which endorses NEI 18-04, Revision 1, with clarifications.

The applicable PDC for the evaluation of the civil and structural analysis for the buildings and structures include:

- PDC 1, “Quality Standards and Records”;
- PDC 2, “Design Bases for Protection Against Natural Phenomena”; and
- PDC 4, “Environmental and Dynamic Effects Design Basis.”

3.7.1 Technical Evaluation

The NRC staff reviewed the analysis and design approaches covered in PSAR section 3.7 that are applicable to the LMGS SR and NSRST structures described in PSAR sections 6.4.1 and 7.3.1. The NRC staff’s review focused on understanding the basis for the selection of initial seismic categories; the selection of codes and standards, including the use of certain codes and standards as a type of special treatment; and the analysis and design criteria for structures, including the conformance of the analysis and design criteria to applicable regulatory guidance. The guidance discussed above provides criteria for analysis and design, including applicable codes and standards, for SR structures. The NRC staff’s review of codes and standards identified for the design and analysis of NSRST SSCs focused primarily on understanding LME’s conformance to the selected codes and standards, how these codes and standards are supplemented by use of endorsed codes and standards (i.e., as a type of special treatment for NSRST SSCs), and how the application of these codes and standards ensures safety and performance of the SSCs.

Seismic Categorization and Seismic Design Basis

The NRC staff reviewed the seismic categorization approach described in PSAR section 3.7.3. The applicant categorized the SSCs as seismic category I (SC-I), seismic category IIA (SC-IIA), seismic category IIB (SC-IIB), or seismic category III (SC-III). The criteria proposed by LME for assigning the initial seismic categories to SSCs considers the safety function of the SSCs, their seismic risk significance, their contribution to adequate DID in response to a seismic event, **[[ ]]** and the potential for seismic interaction.

The applicant stated that SC-I applies to SSCs that perform RSFs that are necessary to shut down the plant during a safe shutdown earthquake (SSE) and maintain the plant in a safe state. For SC-I SSCs, the applicant refers to the SSE described in PSAR section 2.5.2.6.6, as the SC-I design response spectrum (DRS). The NRC staff’s review of the SSE is addressed in

section 2.5.2.6 of this SE. PSAR section 3.7.3.6 states that, at the preliminary design stage, SSCs that perform an RSF (i.e., all SR SSCs) are categorized as SC-I. PSAR tables 3.7-1 and 3.7-2, show the preliminary seismic categorization of structures and systems, respectively, at the preliminary design stage. **[[**

]]. The NRC staff notes that SC-I is the appropriate seismic category assigned to SR SSCs and includes the assignment of the DRS corresponding to the SSE and nuclear codes and standards. The NRC staff finds the preliminary assignment of SC-I to **[**

] to be consistent with the guidance in RG 1.29 and is therefore acceptable.

The applicant described that SC-IIA **[[**

]] seismic risk-significant SSCs, determined consistent with the NEI 18-04 risk metrics; and SSCs necessary for DID adequacy in response to a seismic event, determined consistent with the criteria in table 5-2 of NEI 18-04. PSAR section 3.7.3.6 states that, **[[**

]].

PSAR section 3.7.3.3 describes the DRS for SC-IIA SSCs. SC-IIA SSCs are designed to **[[**

]] Moreover, LME's approach for deriving the scale factors includes compliance with the ASCE 43 seismic design criteria of achieving a 1 percent or smaller frequency of unacceptable performance conditioned on design-basis earthquake shaking and a 10 percent or smaller annual frequency of unacceptable performance conditioned on 150 percent of design-basis earthquake shaking. **[[**

].

As shown in PSAR table 3.7-1, at the preliminary design stage, SC-IIA is assigned to all the NSRST building structures identified in the PSAR. PSAR table 3.7-2 summarizes the seismic categories for LMGS' major systems, and **[[**

]. The NRC staff finds the preliminary assignment of SC-IIA to all NSRST SSCs that **[**

] acceptable because the seismic design bases for these SC-IIA SSCs, including the DRS and analysis/design codes and standards, are consistent with the design bases for SC-I SSCs. Further, the NRC staff finds that LME's approach for determining the **[[**

]].
Also, based on LME's consistency with ASCE 43-19 and its commitment to the performance of a seismic PRA as indicated above, the NRC staff finds that the determination of the final **[[ ]]** can be left for later consideration during the review at the OL application stage.

The applicant stated that SC-IIB **[[ ]]** could seismically challenge SC-I and SC-IIA SSCs (i.e., SSCs with seismic interaction potential with SC-I and SC-IIA SSCs). These SC-IIB SSCs are seismically qualified to prevent adverse interactions with SC-I and SC-IIA SSCs in accordance with ASCE 7-16 and are designed to an enhanced DRS. PSAR section 3.7.3.4 describes the DRS for SC-IIB, which is defined by replacing the ASCE 7-16 design earthquake (i.e. 2/3 of the risk targeted maximum considered earthquake [MCE_R]) with the SC-I DRS **[[ ]]** to achieve a seismic DBE for the SC-IIB SSCs consistent with **[[ ]]**. The definition of seismic demands for SC-IIB SSCs also considers the use of the Risk Category IV **[[ ]]**. The NRC staff's review found the assignment of SC-IIB to SSCs with seismic interaction potential with SC-I and SC-IIA SSCs to be acceptable because the seismic demand for these SSCs is defined based on the most stringent seismic provisions in ASCE 7-16 (i.e. those associated with Risk Category IV) and is defined consistent with **[[ ]]**.

The applicant categorized the rest of the SSCs (i.e., those with no safety-significant seismic design requirements) as SC-III SSCs. For SC-III SSCs, PSAR section 3.7.3.5 describes the use of ASCE 7-16 and the parameters listed in PSAR table 3.7-3 to develop the SC-III DRS. The NRC staff confirmed, via the ASCE Hazard Tool, that the parameters listed in PSAR table 3.7-3 correspond to the parameters for the site coordinates provided in PSAR section 1.1.3 and ASCE Site Class D. Based on geotechnical data for the LMGS site, provided in PSAR chapter 2, the site classification under ASCE 7-16 would be Site Class F. However, during the regulatory audit, the NRC staff reviewed LME's basis for the use of ASCE 7-16 Site Class D in the development of the SC-III DRS. Specifically, the NRC staff reviewed LME's assessment of site-specific properties against the conditions in section 20.3.1 of ASCE 7-16, which allow the use of adjusted Site Class D F_a and F_v values (short-period and long-period site coefficients, respectively) in the development of the DRS for a site that classifies as Site Class F. The NRC staff's review confirmed that LME met the conditions to use the adjusted Site Class D values and therefore finds the SC-III DRS, as described by the parameters in PSAR table 3.7-3, to be acceptable.

Based on the NRC staff's review described above, the NRC staff finds that LME's seismic categorization associated seismic design basis provides reasonable confidence at the CP application stage that safety-significant SSCs will meet their intended function in response to a seismic event. The NRC staff also finds LME's approach to be adequately risk-informed and acceptable because it is consistent with the risk-informed approach described in NEI 18-04, as endorsed by RG 1.233.

Loads and Load Combinations

The NRC staff reviewed PSAR section 3.7.4, which summarizes the postulated loads for the design of structures. The postulated loads are applied to each structure based on the conditions to which each structure may be subjected. The postulated loads include normal, severe environmental, extreme environmental, and abnormal loads. In PSAR section 3.7.5, the applicant described the load combinations associated with different structures and safety classifications. The load combinations listed include load combinations for concrete structures, including the load combinations applicable to SR, NSRST, and NST structures; load combinations for steel structures, including the load combinations applicable to SR, NSRST, and NST structures; load combinations for foundations; and load combinations for construction.

The NRC staff reviewed the load combinations for SR concrete and steel structures and foundations described in PSAR section 3.7.5. The NRC staff finds the information related to the postulated loads for the design of SR structures to be adequate because the postulated loads are consistent with the principal design codes and standards for SR concrete and steel structures, namely American Concrete Institute (ACI) 349 and American Institute of Steel Construction (AISC) N690, and RGs 1.142 and 1.243, and consider the environment in which the LMGS structures are required to function. The NRC staff also verified the load combinations for SR concrete and steel structures are consistent with RGs 1.142 and 1.243, respectively. Therefore, the load combinations for SR structures are acceptable. Additionally, the NRC staff finds that the load combinations for the SR foundation are consistent with the guidance in sections 3.8.4 and 3.8.5 of NUREG-0800 and are therefore acceptable.

The NRC staff reviewed the load combinations for NSRST and NST concrete structures and foundations, and steel structures described in PSAR section 3.7.5, which references ACI 318-14, AISC 360-16, and ASCE 7-16 as the basis for these load combinations. The NRC staff verified that the load combinations used for the NSRST and NST structures are consistent with the referenced codes and standards. Additionally, LME augmented these load combinations with load combinations from nuclear codes and standards (i.e., ACI 349 and AISC N690) to address abnormal loads. The NRC staff finds the load combinations for NSRST and NST structures acceptable because they are consistent with the referenced design codes and standards and are supplemented by SR loads and load combinations as necessary.

The NRC staff reviewed PSAR section 3.7.5.4, which states that construction loads and associated load combinations are considered in accordance with ASCE 37. The consideration of construction loads in accordance with ASCE 37 is consistent with the guidance in NUREG-0800, section 3.8.4, and is therefore acceptable.

Design and Analysis of Structures

The NRC staff reviewed the design and analysis methods for structures as described in PSAR section 3.7.6. These methods include equivalent static analysis, response spectrum analysis, and soil-structure interaction (SSI) analysis. The equivalent static analysis is applied consistent with the provisions in section 4.5 of ASCE 4-16. In equivalent static analysis, a factor of 1.5 is applied to the peak spectral acceleration of the applicable response spectrum. The use of this factor is consistent with both the provisions in section 4.5 of ASCE 4-16 and section 3.7.2 of NUREG-0800. The response spectrum analysis for SR structures is applied consistent with the applicable guidance in section 3.7.2 of NUREG-0800, and RGs 1.61 and 1.92. Per PSAR section 3.7.6.5, the structural analyses for the preliminary design stage were performed in

SAP2000, which is a widely used and recognized software in the nuclear industry. Based on LME's use of these methods in accordance with the provisions of ASCE 4-16, RG 1.61, RG 1.92, and section 3.7.2 of NUREG-0800, the NRC staff finds the equivalent static and response spectrum analyses methods acceptable.

The NRC staff reviewed PSAR section 3.7.6, which states that SSI effects are considered for each building. The consideration of SSI effects for the buildings informs the determination of the seismic demands transmitted to the SSCs housed in and supported by the buildings. LME's assessment of SSI effects is performed in accordance with referenced building codes and standards, including ASCE 4-16, and ASCE 43-19, and includes consideration of the dynamic characteristics of the soil, structure/foundation system, and supported components, and assessment of whether SSI effects are conservatively bounded by preliminary analyses based on fixed-base analysis assumptions. Based on these and additional considerations described in PSAR section 3.7.6, SASSI frequency domain analyses are performed for SR structures and adjacent NSRST structures. The associated finite element models are developed in ANSYS and subsequently transferred to SASSI for the performance of the SSI analyses. The analyses are conducted using the SASSI substructure subtraction approach following the provisions and guidance in ASCE 4-16 and section 3.7.2 of NUREG-0800. Five sets of ground motion time histories are used to perform the analyses and, consistent with section 3.7.2 of NUREG-0800 and as confirmed during the audit, the development of the time histories includes verification of enveloping criteria for the DRS and target power spectral density function. Furthermore, the analyses address the flexibility and impedance of the supporting soils, including consideration of lower bound, best estimate, and upper bound strain compatible soil profiles.

To verify the adequacy of the models and SASSI analysis method, and to ensure accuracy of the results, LME's approach includes validation and verification analyses such as fixed-base harmonic analysis and pseudo-static 1-g analysis performed in both SASSI and ANSYS. The resulting acceleration transfer functions are then compared. These are standard verification techniques and therefore acceptable to the NRC staff. Additionally, LME's approach includes validation of its modeling and analysis approaches, including implementation of the SASSI subtraction method, by performing sensitivity analyses and comparing against SASSI benchmark problems (including ones with deep foundation elements) consistent with the guidelines in ASCE 4-16 and section 3.7.2 of NUREG-0800. Moreover, per PSAR section 3.7.6.5, LME will ensure that vendor-supplied software is compliant with NQA-1.

Based on the NRC staff's review described above, the NRC staff determined that LME's approach for considering SSI effects is adequate because the consideration of SSI effects is performed in accordance with the codes and standards referenced in PSAR section 3.7 and PSAR chapters 6 and 7. Moreover, the NRC staff finds the approach for SASSI frequency domain analyses to be adequate because the analyses are performed and verified consistent with the referenced codes and standards and related guidance including ASCE 4-16 and section 3.7.2 of NUREG-0800. On this basis, the NRC staff concludes that LME's approach for considering SSI effects meets the applicable SSI requirements of Appendix S to 10 CFR Part 50. During the OL application stage, the NRC staff will confirm that the inputs and methodologies used for the analysis and the final design follow the methodology described in the PSAR.

The NRC staff reviewed the additional design and analysis aspects summarized in PSAR subsections 3.7.6.2, 3.7.6.3, and 3.7.6.4 for steel structures, concrete structures, and foundations, respectively.

For steel structures, the PSAR states that **[**

]. The NRC staff notes that ASCE 43-19 is primarily used for SR SSCs and therefore its application to NSRST SSCs represents an enhanced seismic design treatment for these SSCs. Therefore, the use of ASCE 43-19 for NSRST steel structures as described in the PSAR is acceptable.

For concrete structures, PSAR section 3.7.6.3 describes, in part, the calculation of the modulus of elasticity and shear modulus for concrete structures as a function of concrete compressive strength and Poisson's ratio. The PSAR indicates that the concrete compressive strength f_c is a minimum of 5,000 psi for all structures, and the Poisson's ratio is 0.17. The NRC staff finds the definitions of modulus of elasticity, shear modulus, concrete compressive strength, and concrete Poisson's ratio acceptable because they conform with applicable ACI concrete codes, ASCE standards, and American Society of Testing and Materials (ASTM) specifications.

PSAR section 3.7.6.3 also describes the consideration of concrete cracking for seismic load cases for SC-I and SC-IIA/B structures. To account for concrete cracking in the evaluation of seismic load cases, the stiffness reduction factors from table 3-2 of ASCE 4-16 are used. The stiffness reduction factors from table 3-2 of ASCE 4-16 conform to the concrete cracking criteria in section 3.7.2 of NUREG-0800 and are therefore acceptable. Additionally, this PSAR section describes the use of either appendix D of ACI 349-13 or chapter 17 of ACI 318-14 for the design of anchor bolts. The specific code used is based on the applicable performance goal of the anchor being designed. The use of appendix D of ACI 349-13 is acceptable because appendix D of ACI 349-13 is endorsed in RG 1.199. The use of appendix D of ACI 349-13 represents a special treatment for NSRST concrete structures. Based on the enhanced seismic design requirements for SC-IIA NSRST structures, including SSE based DRS, and the use of appendix D of ACI 349-13, where determined necessary, the NRC staff also finds the use of chapter 17 of ACI 318-14 for NSRST concrete structures acceptable.

For foundations, PSAR section 3.7.6.4 describes the use of deep piles or shafts and shallow foundations for building structures. This PSAR section describes geotechnical and structural criteria related to foundation design. This section of the SE focuses on the structural criteria, whereas the geotechnical criteria is evaluated in section 2.5.4.1 of this SE. The PSAR describes the use of minimum safety factors of 1.5 for sliding and overturning for normal loading conditions and 1.1 for extreme environmental loading conditions. These factors of safety conform to the criteria in section 3.8.5 of NUREG-0800 and are therefore acceptable. The PSAR also describes the use of ACI 349-13 for evaluating axial, flexural, and shear stresses for the SR foundation. ACI 349-13 is endorsed by RG 1.142, which as described in PSAR table 1.4-2 is implemented by LME without deviations, and therefore the NRC staff finds that LME's use of ACI 349-13 for the SR foundation is acceptable. For NSRST foundations, the principal concrete code used for the LMGS design is ACI 318-14. As described above, and in the structures related sections in chapter 6 and 7 of this SE, the use of ACI 318-14 for NSRST structures, including their foundations, is supplemented with applicable provisions from ACI 349-13. Further the design of NSRST structures and foundations considers additional special treatments associated with their SC-IIA seismic categorization. Based on the use of ACI 318-14 along with these special treatments, the NRC staff finds the use of ACI 318-14 in the design of the NSRST foundation to be acceptable.

Additionally, PSAR section 3.7.6.4.3 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. For these load cases that include differential settlement loads, 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, LME ensures adequate protection against differential settlement by conducting the foundation related work in conformance with LME's QAP and verifying the capacity of individual piles (where part of the foundation system) by load tests. The NRC staff's review concludes that conformance with the serviceability requirements from the above codes and standards, compliance with applicable quality assurance requirements, and capacity verifications via load test provides reasonable assurance that the foundations will remain functional when subjected to differential settlement loads and associated load combinations.

Seismic Analysis of Systems and Components

The NRC staff reviewed PSAR section 3.7.7.1, which describes the applicant's approach for developing in-structure response spectra (ISRS). The ISRS are obtained from the SSI analyses described above and are used for the seismic design and qualification of systems or equipment supported at various locations within the structures. Two horizontal and one vertical ISRS are obtained at interface locations with supported SSCs. The spectral ordinates of the ISRS are calculated at multiple frequencies, the resulting spectra are smoothed and associated spectral peaks are broadened in accordance with the guidance in RG 1.122 and sections 3.7.1, and 3.7.2 of NUREG-0800. Based on consistency with this regulatory guidance, the NRC staff finds LME's approach for developing ISRS to be acceptable. The NRC staff's review of the final ISRS and its use in equipment and systems design will be performed at the OL application stage.

The NRC staff reviewed PSAR section 3.7.7.2, which describes the seismic qualification methods used for systems and components. These methods include qualification by analysis, testing, similarity, earthquake experience data, or a combination of these methods. The NRC staff finds these methods to be acceptable because they represent standard seismic qualification methods within the scope of the codes and standards, including ASCE 43-19, ASME QME-1, and IEEE 344 and associated regulatory guidance, including RG 1.100, Revision 4. The NRC staff will review LME's implementation of these methods at the OL application stage.

Materials

The NRC staff reviewed PSAR section 3.7.8, which describes the principal materials for structural analysis and design including concrete, reinforcing steel, structural steel, steel bolts, welding electrodes, formed metal decking, grating, checkered plate, and anchor bolts. The NRC staff finds that the applicant's information on principal materials for structural analysis and design are acceptable because they conform to applicable primary design codes, including ACI 349, AISC N690, ACI 318, and AISC 360, referenced standards within the scope of the primary design codes and standards, and ASTM standards. The NRC staff will confirm the use of the materials described in the PSAR in its review of the final design at the OL application stage.

3.7.2 Conclusion

Based on its evaluation above, the NRC staff concludes that the analysis and design criteria and associated methods described in PSAR section 3.7 are acceptable and meet the requirements in PDCs 1, 2, and 4, Appendix S to 10 CFR Part 50, and the requirements of 10 CFR 50.34(a), for the issuance of a CP in accordance with 10 CFR 50.35.

3.8 Structural Evaluation of Metallic Components

This section describes the NRC staff's review of the structural evaluation methods for the SR and NSRST metallic SSCs presented in PSAR section 3.8. PSAR section 3.8 discusses the ASME codes and standards to be used for the structural evaluation along with the development of a material qualification plan to supplement the Code requirements and ensure the structural integrity of all the SR and NSRST metallic SSCs.

The applicable regulatory requirements for the evaluation of the metallic component structural evaluation methodology are as follows:

- 10 CFR 50.34(a)(3)(i);
- 10 CFR 50.34(a)(3)(ii);
- 10 CFR 50.34(a)(4);
- 10 CFR 50.34(a)(8);
- 10 CFR 50.34(a)(12);
- 10 CFR 50.35(a)(3); and
- 10 CFR Part 50, Appendix S.

The applicable guidance for the evaluation of the metallic component structural evaluation methodology is as follows:

- RG 1.87, Revision 3, "Acceptability of ASME Code, Section III, Division 5, "High Temperature Reactors,"" (ML25176A084); and
- RG 1.100, Revision 4, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," (ML19312C677).

The PDCs applicable to the evaluation of the metallic component structural evaluation methodology are as follows:

- PDC 1, "Quality Standards and Records";
- PDC 2, "Design Bases for Protection Against Natural Phenomena";

- PDC 4, “Environmental and Dynamic Effects Design Basis”;
- PDC 14, “Reactor Helium Pressure Boundary”;
- PDC-CDC 15, “Reactor Helium Pressure Boundary Design”;
- PDC 31, “Fracture Prevention of Reactor Helium Pressure Boundary”;
- PDC-RFDC 70, “Reactor Structural Design Basis”; and
- PDC-RFDC 71, “Reactor Support Structure Design Basis.”

3.8.1 Technical Evaluation

The NRC staff reviewed the methods of structural evaluation for the SR and NSRST metallic SSCs in PSAR section 3.8 to determine whether the structural evaluation methods for metallic SSCs are adequate to meet applicable requirements in PDC 1, PDC 2, PDC 4, PDC 14, PDC-CDC 15, PDC 31, PDC-RFDC 70, and PDC-RFDC 71 as described in NRC-approved “Xe-100 Licensing Topical Report Principal Design Criteria” (IBR) along with deviations discussed in PSAR section 5.3.

The NRC staff reviewed PSAR section 3.8.2, which states that for the design of SR SSCs, LME plans to follow the 2023 Edition of ASME BPVC III.5, which is endorsed by RG 1.87, Revision 3, and the related code cases (e.g. N-942, N-861-2, and N-862-2). LME states in the PSAR that it will comply with the exceptions and limitations in RG 1.87, Revision 3. In addition, PSAR section 3.8.2 describes a construction code assessment process to identify the areas that code supplementation is needed. The PSAR states that the construction code assessment process considers material selection, geometric characteristics, and design and service conditions as listed in PSAR table 3.8-2. In accordance with 10 CFR 50.34(a)(8) and 10 CFR 50.35(a)(3), the R&D program described in PSAR section 13.3 (see PSAR table 13.3-1) provides the areas where code supplementation is needed for SR SSCs and includes **[[ ]]** PSAR section 3.8.2 states that the construction code assessment will also evaluate the need to implement conservative modifications to the Code Book requirements to address uncertainties in the Class B allowable stress values for long design lifetimes. The NRC staff finds that the selection of the 2023 Edition of ASME BPVC III.5 and the related code cases for the design of SR SSCs complies with the quality standard requirements of PDC 1 because the selected Code is endorsed in RG 1.87, Revision 3, with exceptions and limitations, as an acceptable approach to assure the mechanical/structural integrity of SSCs for non-LWR license applications. In addition, the NRC staff finds that the construction code assessment method described in PSAR section 3.8.2 provides assurance that the selected Code will be supplemented or modified as necessary to assure a quality product in keeping with the safety-significant function, thus satisfying the corresponding requirements in PDC 1.

The NRC staff reviewed PSAR section 3.8.3, which states that for the design of NSRST SSCs non-nuclear codes and standards such as ASME B31.1 and ASME B31.3 piping codes and ASME Section VIII, Division 1 and Division 2 will be used. PSAR section 3.8.4.1 describes a construction code assessment method to evaluate the applicability and sufficiency of the Code selected for NSRST SSCs. The method considers material selection, geometric characteristics,

and design and service conditions to determine the appropriate construction code approach commensurate with safety significance and safety functions. PSAR section 3.8.4.1 states that, in the case that augmentation of the non-nuclear codes and standards is necessary, the non-nuclear codes selected are supplemented with additional rigor (e.g., use of nuclear codes, enhanced weld integrity measures, and usage of more conservative allowable stress values to cover the design uncertainties) to ensure safety function performance. ASME BPVC Section III, Division 5 is applied where creep-fatigue in the elevated temperature service or other considerations necessitate nuclear-specific requirements for NSRST SSCs. PSAR section 3.8.4 also identifies areas where supplementation to ASME BPVC Section III, Division 5 requirements may be necessary based on the preliminary design of the NSRST SSCs. In addition, in PSAR section 3.8.5.2, LME identified that **[[**

]] and outlined potential methods to address these Code limitations. PSAR section 13.3 states that only the assessments, testing, and/or analysis that comprise code supplementation activities for the SR and risk-significant NSRST SSCs are included in the R&D program. For the preliminary design of the LMGS reactors, the code supplementation activities identified for DID NSRST metallic SSCs are not included in the R&D program, but included in the material qualification plan, which will be implemented in accordance with the design control requirements of the LME quality assurance program description (QAPD). In addition, PSAR section 13.3 states that the LMGS R&D program will be reviewed and updated to include any SSC that is reclassified as SR or risk-significant NSRST as the design matures. The NRC staff finds that the approach described in PSAR section 3.8.4.1 provides a systematic approach to ensure that the construction codes for NSRST SSCs as described in PSAR section 3.8.3, along with the supplementations, as needed, are appropriate and adequate to assure a quality product for the NSRST SSC designs commensurate with their safety significance and safety functions, thus satisfying the requirements of PDC 1. The NRC staff will evaluate the maturation of the R&D program through the required R&D updates as well as the material qualification plan during the OL application review.

In addition, the NRC staff reviewed LME's plan for the functional qualification of active mechanical equipment in PSAR section 3.8.1, which includes: (1) a commitment to full conformance to RG 1.100, Revision 4 for SR equipment and (2) a plan to use a graded approach similar to that in RG 1.100 for NSRST equipment. RG 1.100 endorses ASME QME-1-2017 for the functional qualifications of active mechanical equipment, and the NRC staff finds that LME's approach for the functional qualification of active mechanical equipment complies with the requirements of PDC 1 for the use of quality standards by following ASME QME-1-2017 and RG 1.100. The NRC staff will review the graded approach for the qualification of NSRST active mechanical equipment during the OL application review.

Based on the above information, the NRC staff finds that the applicant's structural evaluation methods for metallic structures meet the requirement of PDC 1 for both SR and NSRST SSCs.

The NRC staff reviewed PSAR section 3.8.1, which states that the deflections and accelerations at component supports and interfaces resulting from seismic events, as determined in PSAR section 3.7, are considered in the stress analysis in accordance with 10 CFR Part 50, Appendix S. LME discussed the methods for seismic qualification of active mechanical equipment in PSAR section 3.8.1, which includes: 1) a commitment to full conformance to RG 1.100, Revision 4 and QME-1 for SR SSCs; 2) a plan to use a graded approach similar to

that in RG 1.100 and QME-1 for risk-significant NSRST SSCs that will be identified as special treatments as part of the OL application; and 3) a plan to use a graded approach similar to RG 1.00 for NSRST SSCs for DID but only for AOO and operating basis earthquake level seismic events. The NRC staff finds that these seismic qualification methods for active mechanical equipment are acceptable because they align with RG 1.100 and the general seismic evaluation approach in PSAR section 3.7, which complies with PDC 2. In addition, PSAR section 3.8 indicates that the LMGS design will comply with the ASME codes (i.e., ASME BPVC III.5, B31.1 and B31.3 piping codes, and ASME Section VIIII, Division 1 and Division 2), which all require that earthquake loads be considered as a dynamic load in the design of an SSC.

The NRC staff finds that the LME structural evaluation method of metallic structures meets the requirement of PDC 2, with review of the graded approach similar to RG 1.100 for seismic qualification of NSRST SSCs reasonably being left to the OL application as discussed above.

The NRC staff reviewed PSAR section 3.8.2.1 and 3.8.4.1, which discussed the environmental compatibility assessment approach for SR and NSRST SSCs. The PSAR states that the environmental compatibility assessment methods will identify environmental conditions that could degrade the SSCs over time (e.g., corrosion, radiation, temperature, chemical exposure) and evaluate whether the SSC's materials and design features are compatible with those conditions. These methods will also determine applicable degradation modes and mitigate the applicable degradation modes through the development of strategies, such as reducing allowable stress values to account for the effects of degradation during design and implementing inservice inspection and health monitoring during operation. PSAR sections 3.8.2.2 and 3.8.4.2 state that test proposals will be developed for SR and NSRST SSCs to support confirmatory testing for material compatibility in reactor-grade helium, justification for determining non-applicable degradation modes, and the strategy for mitigating applicable degradation modes. LME will provide the details of the environmental compatibility assessments for SR and NSRST metallic SSCs in the LMGS material qualification plan, which will be implemented in accordance with the requirements of the LME QAPD. Further, the assessments for SR metallic SSCs are included in the LMGS R&D program described in PSAR chapter 13 to resolve safety questions related to performance of these SSCs in the expected Xe-100 environment operating conditions. PSAR table 13.3-1 provides details on qualification assessments, testing, and analysis to be performed within the R&D program for the metallic SR SSCs. PSAR section 13.3 states that only the assessments, testing, and/or analysis that comprise environmental compatibility assessment activities for the SR and risk-significant NSRST SSCs are included in the R&D program. For the preliminary design of the LMGS reactors, the environmental compatibility assessment activities identified for DID NSRST metallic SSCs are not included in the R&D program, but included in the material qualification plan, which will be implemented in accordance with the design control requirements of the LME QAPD. In addition, PSAR section 13.3 states that the LMGS R&D program will be reviewed and updated to include any SSC that is reclassified as SR or risk-significant NSRST as the design matures.

Based on the discussion above, the NRC staff finds that LME's environmental compatibility assessment approach, along with the planned testing, provides a comprehensive approach to systematically consider the impact of operating environments and service degradation of the safety-significant SSCs and thus satisfies the corresponding requirements in PDC 4. The NRC staff will evaluate the maturation of the R&D program through the required R&D updates as well as the material qualification plan at the OL application stage.

As discussed earlier in this section, the NRC staff finds that by using quality codes and standards with appropriate supplementation, sufficiently considering the environmental compatibility and service degradation, and designing the SSCs to withstand the effects of earthquakes, the structural evaluation method of metallic SSCs described in PSAR section 3.8 satisfies the requirements of PDCs 1, 2, and 4. On this basis, the NRC staff determined that the relevant requirements of PDC 14, PDC-CDC 15, PDC-RFDC-70, and PDC-RFDC-71 are met as described below. The NRC staff determined that PDC 14 is met because the structural evaluation method supports the structural integrity of safety-significant metallic SSCs composing the HPB for the reactor HPB to be designed, fabricated, and erected so as to have an extremely low probability of abnormal leakage and unacceptable moisture ingress. The NRC staff determined that PDC-CDC 15 is met because the structure evaluation method supports the structural integrity of safety-significant metallic SSCs composing the HPB for safety-significant SSCs that are part of the HPB to be designed with sufficient margins to ensure that the design conditions of the HPB are not exceeded during normal operation and AOOs. The NRC staff determined that PDC-RFDC 70 is met because the structure evaluation method supports the structural integrity of safety-significant metallic SSCs such that a reactor system capable of affecting core geometry is designed such that core integrity is maintained and there is a low probability of rapidly propagating failure during DBEs and DBAs. The NRC staff determined that PDC-RFDC 71 is met because the structure evaluation method supports the structural integrity of metallic reactor support structures for these structures to protect the reactor vessel and reactor system geometry during DBEs and DBAs.

As discussed earlier in this section, PSAR sections 3.8.2.1 and 3.8.4.1 describe the construction code and environmental compatibility assessment methods to consider the service conditions to evaluate the adequacy of the selected Codes for the design and construction of metallic SR and NSRST SSCs. For SR SSCs, the PSAR states that LME will use ASME BPVC III.5, which considers service degradation of material properties, fatigue, creep, and stress rupture at elevated temperatures and provides an acceptable approach for the design and construction of non-LWRs per RG 1.87, Revision 3. For NSRST SSCs, PSAR section 3.8.4.1 describes that LME will supplement the code selected by considering the service conditions and failure modes. For both SR and NSRST SSCs, the PSAR states that environmental compatibility assessments consider the service conditions and degradation of the material properties. Based on this information, the NRC staff finds that the structural integrity evaluation method in PSAR section 3.8 meets the relevant requirement in PDC 31 to consider service conditions, degradation of material properties, and failure modes.

The NRC staff reviewed PSAR sections 3.8.1.1 and 3.8.3.1, which describe that, for both SR and NSRST SSCs, data obtained from the material qualification plan will be used as a basis for adjusting design properties to account for service effects that could reduce code-established margins. The level of uncertainty in the material properties and the capability to verify design assumptions are considered in the degree of conservatism applied to reasonably maintain code-established margins. The NRC staff finds that this consideration of uncertainties in material properties and the effects of service conditions on material properties satisfies the PDC-31 requirements for considering uncertainties in material properties. The assessment of compliance to the PDC-31 requirement for the consideration of uncertainties in stresses and size of flaws in the reactor HPB design is performed in PSAR chapters 6 and 7.

Based on the information above, the NRC staff finds that the LME structural evaluation method of metallic structures meets the requirements of PDC-31 for the CP application stage.

Based on its review, the NRC staff finds that the structural evaluation methodology for safety-significant metallic SSCs described in PSAR section 3.8 provides a reasonable approach to ensuring the structural integrity of SR and NSRST SSCs for different service conditions with applicable potential material degradation considered

3.8.2 Conclusion

Based on its review, the NRC staff finds that the preliminary information on the structural evaluation methodology for safety-significant metallic SSCs described in PSAR section 3.8 is consistent with the applicable regulatory guidance and meets the regulatory requirements in 10 CFR 50.34 and 10 CFR 50.35. Specifically, the information provided:

- meets the requirements of 10 CFR 50.34(a)(3)(i) by describing the preliminary design identifying the applicable PDCs,

- meets the requirements of 10 CFR 50.34(a)(3)(ii) by describing how the metallic SSCs satisfy the applicable PDCs,

- meets the requirements of 10 CFR 50.34(a)(8) and 10 CFR 50.35(a)(3) by identifying the R&D needs to confirm the performance of the SR metallic materials in the Xe-100 environmental operating conditions, and

- meets the requirements of 10 CFR 50.34(a)(12) by identifying PDC 2 requirements pertaining to the LMGS seismic evaluation approach to comply with the earthquake engineering criteria in 10 CFR Part 50, Appendix S.

Further, as discussed above, certain details of the structural evaluation methodology for safety-significant metallic SSCs are not yet available because they depend on the completion of ongoing confirmatory research, testing, and development identified in accordance with 10 CFR 50.34(a)(8). Consistent with 10 CFR 50.35(a)(3), the NRC staff finds that the applicant has identified activities consistent with the preliminary design to confirm the performance of the SR metallic materials and that these activities are planned to be completed in time to support the OL application. Therefore, the NRC staff concludes that the absence of this information at the CP stage does not preclude issuance of a CP. The NRC staff will review the remaining structural evaluation information, including the results of the R&D program and material qualification plan, during the OL application review.

3.9 Excore Criticality Analyses

PSAR section 3.9 describes criticality safety for areas on the site containing nuclear fuel located outside of the reactor These areas include

- Spent Fuel Intermediate Storage Facility (SFISF)
- Fuel Handling Auxiliary Building (FHAB)
- Fuel Canister Interim Storage, and
- Fuel Handling System (FHS)

PSAR section 3.9 is focused on the methodologies used to determine the upper safety limit (USL) and perform criticality analyses for each area where new and spent fuel is stored and

handled. The preliminary designs for these areas and their SSCs are provided in PSAR section 7.3 and evaluated in the associated section of this SE.

An exemption to 10 CFR 50.68(b)(7) is requested in LME's CP Application, Part V Non-Applicabilities and Exemptions (ML25090A065), and evaluated in chapter 15 of this SE.

The applicable regulatory requirements for the evaluation of excore criticality analyses are as follows:

- 10 CFR 50.68, "Criticality accident requirements."

The PDC applicable to the evaluation of excore criticality analyses is

- PDC 62, "Prevention of Criticality in Fuel Storage and Handling."

The applicable guidance for the evaluation of the excore criticality analyses are as follows:

- NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," (ML050250061);
- NUREG/CR-7308, "Sensitivity/Uncertainty Methods for Nuclear Criticality Safety Validation," (ML25099A002);
- NUREG/CR-7311, "Determination of Bias and Bias Uncertainty for Criticality Safety Computational Methods," (ML25120A337);
- ANSI/ANS-8.1-2014 (R2023) "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactor"; and
- ANSI/ANS-8.24-2017 (R2023) "Validation of Neutron Transport Methods for Nuclear Criticality."

3.9.1 Technical Evaluation

The NRC staff reviewed PSAR section 3.9 to assess compliance with the regulatory requirements listed in section 3.9.2. The NRC staff focused its review on the applicant's criticality safety analysis methodology and the preliminary results for criticality safety in the SFISF, the FHAB, fuel canister interim storage, and FHS.

3.9.3.1 Methodology

The NRC staff finds that the applicant is using the KENO module within the SCALE code package to perform excore criticality safety calculations. SCALE, developed and maintained by ORNL, is a comprehensive modeling and simulation suite widely used for nuclear safety analysis and design. SCALE provides tools for conducting criticality safety, radiation shielding, reactor physics, and spent fuel analysis. SCALE offers a broad spectrum of capabilities for nuclear science and engineering, including criticality safety analyses. Specifically,

- CSAS6 supports 3D multigroup and continuous energy eigenvalue Monte Carlo analysis for criticality safety.

- KENO-VI provides advanced geometry modeling and physics capabilities suitable for this application.

Within SCALE, KENO is a Monte Carlo-based code designed for criticality SEs. For the PSAR evaluation, the applicant employed the Control Module for Enhanced Criticality Analysis CSAS6 in conjunction with KENO-VI to perform the criticality assessment. This approach included generating radiation sources and determining the radiological characteristics of the fuel pebbles using ENDF/B-VII.1 continuous energy cross-section nuclear libraries.

The NRC staff reviewed LME's excore criticality analysis calculations and supporting documentation as part of the audit. LME clarified that the SCALE/KENO code packages are maintained and executed under the X-energy QA program, and that the commercial grade dedication (CGD) of SCALE/KENO is currently in progress. LME further confirmed that SCALE 6.2.4 was used for the PSAR preliminary analysis, while SCALE 6.3 - currently undergoing CGD - will be validated and used for the FSAR analysis.

Based on the code features and LME's approach for the code configuration control and CGD, the NRC staff finds the use of the SCALE/KENO software package acceptable for performing the criticality safety analysis.

Excore Criticality Uncertainty Analysis Methodology

PSAR section 3.9.2.2 describes the applicant's approach to calculate the USL, which the NRC staff finds is consistent to the method outlined in NUREG/CR-6698. The applicant states that the USL will be calculated using the equation in PSAR section 3.9.2.2, which includes the bias, bias uncertainty, minimum subcritical margin, and additional subcritical margin that may be needed to account for extensions to the area of applicability (AoA).

PSAR section 3.9.2.2 states that a minimum subcritical margin of 0.05 is typically applied, which is consistent with the requirements of 10 CFR 50.68(b)(2) and (b)(4). This section also notes that the bias is set to zero when the calculated average k_{eff} is greater than 1.0, indicating that the code is overpredicting k_{eff} on average. Setting a positive bias to zero in this case is considered conservative and aligns with the methodology described in NUREG/CR-6698.

NUREG/CR-6698, NUREG/CR-7308, and NUREG/CR-7311 describe the recommended approach for determining bias and bias uncertainty in the USL through validation of the codes and models used in the criticality safety analysis against applicable experimental data. This validation process typically involves:

- Selection of critical benchmarks and critical experimental data representative of the system configuration and parametric range (materials, conditions, and operating parameters) of the analyzed system.

- Evaluation of bias and bias uncertainty based on statistical comparison of calculated and experimental results.

LME assumes a USL of 0.95 for the results presented in the PSAR. As described in PSAR sections 3.9 and 3.9.2.2, LME plans to perform additional studies, including benchmark analyses and bias and uncertainty quantification, using applicable experimental data consistent

with the guidance in NUREG/CR-6698. The PSAR states that these analyses would provide an updated USL value for the FSAR, which is anticipated to be lower than the 0.95 value assumed for the preliminary analyses. for the preliminary analyses.

The NRC staff also reviewed LME's excore criticality analysis methodology plan (Document ID number 005272, "Ex-Core Criticality Methodology and Analysis Plan") during the audit and determined that the proposed approach for selecting experimental data, establishing the acceptable AoA, and performing the calculations of bias, bias uncertainty, USL, and associated statistical evaluations is consistent with NUREG/CR-6698 and applicable ANS standards.

3.9.3.2 Criticality Safety Analyses

The results for the preliminary criticality safety analyses of the SFISF, fuel canister interim storage, FHAB, and FHS are documented in PSAR sections 3.9.3 through 3.9.7, respectively.

Criticality Safety Analysis of SFISF

All four SFISFs are shared between the four modules. Each SFISF contains 660 storage positions arranged across 10 bays. The SFISF is used to store spent fuel canisters (SFCs) and used fuel canisters (UFCs). Fuel stored in UFCs may be returned to the reactor, with typical operation involving six passes through the core.

For the SFISF criticality safety analysis, the applicant makes the following key assumptions:

- One full core upload occupies 45 storage positions consisting of UFCs with fuel pebbles at the maximum licensed U-235 enrichment of 15.5 wt percent.
- The remaining 615 storage positions are occupied by SFCs containing fuel pebbles that have undergone a single pass through the core which is conservative since fuel pebbles typically passes through the core six times.
- The SFISF is assumed conservatively to be flooded inside and outside the canisters with unborated water over a range of water densities, which is a well-established conservative assumption for criticality analysis.
- The most limiting TRISO particle and pebble geometry is used including:
    - A maximum pebble packing fraction 63 percent in canister, based on computational fluid dynamics discrete element analysis.
    - Use of 95th percentile values for pebble and TRISO particle specifications (PSAR table 3.9-1).

These assumptions maximize the amount of fissile material present and the systems reactivity. The NRC staff finds these assumptions appropriately conservative for the criticality analysis. The analysis also assumes an administrative control (see PSAR table 3.9-5) that prohibits stacking of canisters within the SFISF.

LME calculates k-effective for five different spatial arrangements of fresh fuel canisters, as exhibited in PSAR figure 3.9-4. Reflective boundary conditions are applied on the left and bottom boundaries (top view of SFISF), while concrete wall boundary conditions are applied on the remaining four sides. The NRC staff finds that these boundary conditions are conservative because they minimize neutron leakage.

To identify most reactive conditions, the PSAR analysis evaluates 121 combinations of water density inside and outside the canisters. The results demonstrate that k-effective remains below 0.95 for the configuration with up to six fresh fuel canisters per SFISF bay.

Based on these results, the applicant proposes an administrative control (see PSAR table 3.9-5) limiting each SFISF bay to a maximum of six fresh fuel canisters. The NRC staff finds this evaluation acceptable because it demonstrates that the combined preliminary design features and administrative controls maintain the SFISF subcritical under assumed worst-case flooding and moderation scenarios.

Criticality Safety Analysis of Fuel Canister Interim Storage and FHS

In PSAR section 3.9.4, the applicant evaluates the criticality safety of UFCs that are offloaded from the reactor and temporarily stored in the interim storage area prior to transfer to the SFISF. The analysis considers four UFCs arranged in two configurations: (1) all four canisters in direct contact, and (2) canisters separated by 40 cm. To maximize system reactivity, the analysis assumes fresh fuel with 15.5 wt percent U-235 enrichment. Consistent with the methodology applied to the SFISF analysis, LME evaluates 121 cases spanning a range of water densities inside and outside the canisters.

The results demonstrate that the configuration with a minimum 40 cm separation maintains k-effective below the acceptable limit under maximum moderation conditions. Based on these findings, LME proposes administrative control in PSAR table 3.9-5 limiting the interim storage area to no more than four UFCs with a minimum spacing of 40 cm.

For the FHS, PSAR section 3.9.6 states that fewer than 2000 fuel pebbles are present within the system at any time. The interim storage criticality safety analysis shows that a single UFC containing more than 6000 fresh fuel pebbles remains subcritical under all evaluated moderation conditions. Therefore, LME concludes that even a complete spill of the FHS contents would not result in critical configuration under any credible flooding or moderation scenario.

The NRC staff reviewed the methodology and assumptions used for interim storage area criticality analysis, including the bounding fuel enrichment, water density variations, and geometric configurations. The NRC staff finds that the evaluation adequately demonstrates that the proposed administrative controls (see PSAR table 3.9-5) ensure that UFCs in the interim storage area remain subcritical under maximum fissile mass and worst-case moderator conditions. Therefore, the NRC staff concludes that the analysis provides reasonable assurance of subcriticality for the interim storage configuration.

Criticality Safety Analysis of FHAB

PSAR section 3.9.5 evaluates criticality safety in the FHAB under the most limiting conditions of flooding with unborated water. The FHAB stores up to a maximum of 56 fresh fuel transport

packages (FFTP). The analysis assumes that the FFTPs may be stacked up to two layers high. The EMs two conservative configurations which also defines the proposed administrative control (see PSAR table 3.9-5) for the FHAB: one layer of 56 FFTPs and two stacked layers of 28 FFTPs each. PSAR table 3.9-3 provides important parameters of FFTP.

Using the SCALE software package and the FULCRUM interface, the analysis develops simplified FHAB models with concrete walls on four sides to conservatively account for neutron reflection. For additional conservatism, the analysis arranges FFTPs in a hexagonal cluster and positions them in a corner to maximize neutron reflection and minimize leakage, which is a conservative configuration assumption for the criticality analysis.

In PSAR section 3.9.5.2, the analysis specifies fuel pebble parameters to maximize system reactivity. The analysis assumes fresh fuel pebbles with enrichment at 15.5 percent U-235 within UCO kernels and selects fuel parameters (PSAR table 3.9-4) and FFTP geometry to maximize uranium loading including the assumption of whole pebbles for the patrial pebbles touching the FFTP insider surface in the SCALE model. To capture the most reactive conditions, the analysis evaluates water ingress scenarios across a range of moderation conditions. As described in PSAR section 3.9.5.3, the analysis calculates the effective neutron multiplication factor (k-effective) for eleven external water densities (0 percent to 100 percent in 10 percent increments) and 11 internal water densities within the containers, resulting in 121 cases. The results show that k-effective remains below 0.95 at a 95 percent probability and 95 percent confidence level for all scenarios (see PSAR figure 3.9-11). This evaluation demonstrates that the FHAB design and administrative controls (see PSAR table 3.9-5) maintain subcritical conditions under worst-case flooding and moderation scenarios.

Criticality Safety in Other Plant Locations

PSAR section 3.9.7 evaluates criticality safety for normal and postulated off-normal excore fuel pebble handling or storage scenarios, including a postulated single canister breach resulting in a spill of fuel pebbles. The PSAR states that the fully loaded canister configurations - consisting of fresh, spent, and used fuel pebbles analyzed for the SFISF, FHAB, and interim storage - bound all credible fuel storage or handling canister configurations. The NRC staff finds this assumption acceptable because the applicant's analysis encompasses all plausible canister storage locations identified in the preliminary design.

Additionally, the PSAR concludes that the random pebble geometry resulting from a postulated canister breach would not produce a more reactive configuration than those analyzed in the PSAR. The NRC staff finds this conclusion acceptable because the applicant's criticality analysis assumes a bounding pebble packing fraction within the canister. This yields a conservative maximum fissile-material density per unit volume that adequately bounds any random, credible spilled-pebble arrangement.

Based on the applicant's analysis, assumptions, and administrative controls (see PSAR table 3.9-5) described above, the NRC staff determined that the preliminary criticality safety analyses for the SFISF, interim storage, FHAB and FHS:

- demonstrates that the effective neutron multiplication factor (k-effective) does not exceed 0.95 at a 95 percent probability and 95 percent confidence level under the most limiting condition of flooding with unborated water, as required by 10 CFR 50.68(b)(2);

- accounts for optimum moderation scenarios and confirms that k-effective remains below regulatory limits, consistent with 10 CFR 50.68(b)(3); and

- ensures subcriticality under all credible conditions with conservative assumptions regarding fuel enrichment, geometry, and water ingress.

3.9.2 Conclusion

The NRC staff has determined that the excore criticality safety methodology described in PSAR section 3.9 provides a reasonable approach for the criticality safety analyses to satisfy PDC 62 and provides reasonable assurance that criticality will be prevented. The NRC staff determined that the preliminary evaluations described in PSAR section 3.9 provides an adequate analysis and evaluation of the preliminary design and performance of SSCs at the facility per 10 CFR 50.68.

3.10 Non-Core Related Source Radiological Analyses

PSAR section 3.10, “Non-Core Related Source Radiological Analyses,” contains the identification of non-core radionuclide sources that are evaluated through risk-informed supplemental analyses for the CP application. It contains acceptance criteria and preliminary results for analyses associated with radiological sources in the FHS, the spent fuel storage system (SFSS) canisters (both in the handling phase and in the SFISF), the radwaste treatment building, the HSS, and the neutron start-up source.

The applicable regulatory requirements for non-core radionuclide sources are as follows:

- 10 CFR 50.34(a)(1)(ii)(D); and

- 10 CFR 50.34(a)(4).

3.10.1 Technical Evaluation

The NRC staff evaluated the information in PSAR section 3.10 against the following criteria for LBE evaluations in a CP application provided in RG 1.253, section C.4:

1. The design will provide sufficient safety margins during normal operations and transient conditions.

2. The applicant has identified the SSCs necessary to prevent accidents and mitigate accident consequences.

3. The applicant has demonstrated an understanding of the uncertainty associated with the performance of SSCs necessary to prevent accidents and mitigate accident consequences.

3.10.1.1 Methodology

Identification and Grouping of Radiological Sources

PSAR section 3.10.1.1, "Identification and Grouping," states that: "Radiological sources were identified by evaluating each system in the LMGS system breakdown structure to determine if the system contains radionuclides before initiation of an event." PSAR table 3.10-1, "Summary of LMGS Radiological Sources and Evaluations," lists the radiological sources at the LMGS. The NRC staff evaluated the identification of radionuclide sources, in part, by comparing the locations identified in PSAR table 3.10-1 to the system descriptions provided in PSAR chapter 1, "General Plant and Site Description and Overview of Licensing Basis," PSAR chapter 6 and 7, "Non-Safety-Related with Special Treatment Structures, Systems, and Components Criteria and Capabilities." Based on this comparison, the NRC staff finds reasonable assurance that the applicant adequately identified radiological sources for preliminary analysis because: (1) the evaluation through a system breakdown structure, as described in PSAR section 3.10.1.1, is an accepted structured approach to radiological source identification, and (2) the locations identified in PSAR table 3.10-1 are consistent with those locations that contain materials that have been exposed to the neutron flux from the reactor core.

Graded Evaluation Method

PSAR section 3.10.1.2 states that the evaluation of each identified radionuclide source uses the consequence metric of NEI 18-04 (i.e., the 30-day TEDE to a member of the public at the EAB) for comparison against the F-C target and that radionuclides were grouped for evaluation based on similarities in form and the barriers present to retain radionuclides. The evaluation of fuel sources in the FHS and the SFSS located in the SFISF use conservative but not bounding consequence analyses of events. The applicant states that these fuel analyses envelop the risk of challenges experienced by the fuel such that future inclusion of these radionuclide sources in the PRA will provide a lower and more realistic quantification of risk for the FSAR. Alternatively, evaluations of non-fuel radionuclide sources use analyses reflecting the bounding expected radionuclide inventory and credible maximum release fractions. Dispersion factors are quantified by the same methodology used for DBAs in PSAR section 3.6, but releases from the SFISF, radwaste building, and helium services facility use higher dispersion factors because these release points are closer to the dose receptor than the release points for other sources.

The NRC staff reviewed the general evaluation methods to develop MST and dispersion factors in sections 3.1.4 and 3.1.5 of this SE, respectively. The NRC staff's evaluation of the specific isotope inventories is provided in section 3.1.4.1.4 of this SE for inventories listed in PSAR table 3.10-4 for a single fuel pebble, table 3.10-5 for liquid radwaste, table 3.10-6 for solid radwaste, table 3.10-7 for the HPS dust filter, and table 3.10-8 for the low temperature absorbers. For thermal events affecting the fuel in the FHS and SFSS, the source term is calculated considering the effect of oxidation on the fuel pebbles. As documented in Section 6.2.2 of IAEA TECDOC 1674, 2012, the oxidation rate of graphite, which surrounds the fuel pebbles, is very small below 600°C and, thus, the release from affected fuel pebbles in air environment would be very small. Therefore, the NRC staff concluded that peak fuel temperatures providing significant margin to 600°C result in an insignificant offsite dose. The NRC staff evaluated the specific methods, margins, safety features, and results for each non-core related source in sections 3.10.3.2 through 3.10.3.7, which follow. The NRC staff will evaluate the results of the PRA-supported analyses in the FSAR at the OL application stage.

3.10.1.2 Fuel in the Fuel Handling System

The NRC staff reviewed PSAR section 3.10.2, which describes the evaluations of thermal and mechanical events affecting fuel in the FHS. The applicant states that the fuel enters the FHS after approximately 7 days of residence in the defuel chute, allowing pebbles to cool below 250°C before entering the FHS. The applicant states that the FHS includes irradiated fuel pebbles in several parts of the system during normal operation at full power, and the number of pebbles within the system for other operating modes is lower. Therefore, the inventory of radionuclides within the system during normal operation is representative or bounding with respect to other operating modes. The potential source term in the FHS includes the radionuclides in approximately 500 pebbles, which is the number of pebbles the system is designed to accommodate at one time.

Thermal Events Affecting Fuel in the FHS

The NRC staff reviewed PSAR section 3.10.2.1, in which the applicant describes that thermal events are not challenging because the FHS contains a limited number of fuel pebbles, the decay heat from each pebble is small, the heat transfer from the FHS components to the surroundings is effective, and the fuel has high thermal capacity. Therefore, the applicant concluded that specific heat removal mechanisms are not necessary to ensure radionuclide retention for fuel within the FHS. As listed in PSAR table 3.10-8, the applicant determined that the representative thermal event for fuel in the FHS would result in a dose of less than 2.5 mrem (0.025 mSv) TEDE, which is considered acceptable consequences at any frequency.

The NRC staff evaluated the conditions assumed for the FHS representative thermal event. PSAR table 3.10-2 indicated that fuel pebbles within the FHS would have a decay heat generation rate at the start of an FHS transient of 2 watts (W) per pebble. The NRC staff reviewed in the regulatory audit the applicants decay heat model and independently estimated an average decay heat generation rate for fuel that had been operating at the average power per pebble of the Xe-100 for an extended period of time with 7 days decay to confirm that the 2 W value stated in the PSAR was reasonable. The NRC staff estimated the average decay heat using a value of 0.003 W decay heat per W of operating power for thermal fission of low enriched uranium light water reactor fuel with 7 days decay time from section 9.2.5 of NUREG-0800 (ML052350549) and multiplying that value by an average power per pebble determined by dividing the estimated thermal power per module of 200 MWt by 220,000 pebbles. The NRC staff finds that the applicant's stated value of 2 W per pebble, compared favorably with a conservative staff-estimated value of 2.7 W. The lower maximum decay heat per Xe-100 pebble is expected considering each Xe-100 pebble makes several random passes through the reactor to reach full burnup, which results in a longer period of time to reach full burnup and a lower inventory of intermediate-lived isotopes. PSAR section 3.10.2.2 describes that the fuel (and associated heat) is distributed throughout the FHS with concentrations of no more than about 200 pebbles in any one location. As described in PSAR section 7.3.6.1.1, the FHS is primarily housed within the FHAB, with portions in the SST and the canister processing facility (CPF), and the FHS pressure boundary provides direct heat transfer to the surroundings. As described in PSAR section 7.3.10, these structures are normally provided conditioned air by the NST primary loop of the Nuclear island HVAC (NIHV). During transient conditions, NIHV cooling may be lost. However, effective natural circulation heat transfer from FHS components to the surroundings would remain and the necessary heat removal from individual sections of the FHS would be very small (i.e., <400 W). The fuel pebbles retain radionuclides to temperatures much higher than the discharge temperature of 250°C without significant oxidation. Considering the

foregoing, the NRC staff concluded that the preliminary FHS design provides sufficient safety margins for potential thermal challenges during normal operations and transient conditions.

Mechanical Challenges Affecting Fuel in the FHS

The NRC staff reviewed PSAR section 3.10.2.2, in which the applicant describes the analyses performed to evaluate representative mechanical challenges to the fuel within the FHS. The applicant evaluated the potential loss of conveyance gas flow resulting in a drop if a single pebble within an intact FHS HPB and a major postulated failure of the FHS pressure boundary at a location with the highest number of pebbles that could be ejected through the break. The applicant concluded that the internal loss of conveyance gas may cause mechanical damage to the pebble graphite matrix but would not be expected to affect the integrity of the fuel particles within the pebble. The applicant determined that the result would be an insignificant release within the FHS pressure boundary. Therefore, the loss of conveyance gas does not challenge the retention of radionuclides.

As described in PSAR section 3.10.2.2, the applicant also evaluated the consequences of a full break in the FHS conveyance piping allowing for the ejection of pebbles from the FHS at full reactor pressure. The applicant presented the assumed radionuclide inventory of a single pebble in PSAR table 3.10-4. The applicant assumed a conveyance pipe rupture in the discharge lock buffer area, which results in the maximum potential number of affected pebbles. The applicant assumed each of the 206 ejected pebbles experiences a full diameter planar fissure that damages 2.6 percent of the TRISO particles within each pebble and considered AGR data indicating cracked fuel compacts experience 0.74 percent to 3.5 percent damaged TRISO particles. Assuming release of 100 percent noble gases and volatile fission products and a small fraction of particulate radionuclides with no retention in the FHS or surrounding structures, the applicant determined a dose consequence of 5.3 rem (0.053 Sv)TEDE for this conservative representative mechanical event. Preliminary HPB rupture frequency estimates for an FHS full diameter break of 1.19 E-05 per year indicate the event is within the BDBE range. The applicant stated that mechanical impact events would be further evaluated in the PRA supporting the FSAR for a future OL application.

The NRC staff reviewed the applicant's development of the representative mechanical event. The NRC staff considered the design features and special treatments applied to the NSRST FHS design as specified in PSAR section 7.3.6.1, including the seismic design of FHS SSCs to seismic category IIA as described in PSAR section 3.7. The NRC staff finds that the design of the FHS provided appropriate protection of the fuel during normal operating conditions and expected challenges, consistent with the FHS design criteria. For the unlikely and conservative analysis of many damaged pebbles, the NRC staff evaluated the assumptions and methodologies used in assessing the event, including the evaluation of the detailed analysis in the regulatory audit. The NRC staff's evaluation of the radionuclide inventory in each pebble is discussed in section 3.1.4.1.4 of this SE, and the NRC staff finds the developed inventory listed in PSAR table 3.10-4, "Single Pebble Isotope Inventory," reasonable. The NRC staff finds the assumed number of affected pebbles conservative because the FHS contains that number of pebbles in only one location and many other potential break locations would result in a much smaller number of ejected pebbles. The NRC staff concluded that the application of this damage to all affected pebbles results in an overall estimate of damage to TRISO particles that is conservative, recognizing that the ejection of multiple pebbles would result in varying damage states. The estimate of radionuclide release from each damaged particle is also reasonable and based on operating experience and test data for similar fuel structures. Therefore, the NRC staff

concluded that the FHS design will provide sufficient safety margins during normal operations and transient conditions by providing appropriate protection from damage during normal operations and assuring the frequency of more challenging events that could result in appreciable offsite consequences are acceptably low. In addition, the NRC staff finds that the preliminary analysis of the representative mechanical event provides reasonable assurance that more detailed further evaluations of the frequency and consequences of such an event would not substantially reduce the available margin.

3.10.1.3 Non-Fuel Sources in the FHS

The NRC staff reviewed PSAR section 3.10.3, in which the applicant describes its evaluation of non-fuel sources within the FHS. The applicant assessed potential releases from the FHS dust filter and from circulating activity within the FHS. The applicant states that a conservative deterministic analysis of a release from the FHS dust filter shows that this radiological source will likely screen out due to insignificant consequences. The scoping calculation does not credit any physical barriers. Similarly, the release of circulating activity from within the portion of the FHS that can be isolated from the reactor HPB is insignificant (i.e., < 2.5 mrem (0.0025 rem)). The applicant states that a release of circulating activity from the reactor HPB, which includes activity from the non-isolable portions of the FHS, is evaluated as an LBE and, therefore, is outside the scope of this non-core related source evaluation.

The NRC staff considered the significance of these potential non-fuel release sources. The NRC staff noted that PSAR section 7.3.6.1.2 describes that FHS graphite dust is continuously removed by the conveyance gas system and collected in a pulse filter unit and, separately, dust and gas removed from the FHS during process operations (e.g., gas change-over at airlocks and vacuum purge) is routed to the helium recovery potion of the HSS for filtering prior to reuse or release. PSAR section 3.10.6 addresses release of radionuclides contained in the HSS, including graphite dust and radionuclides captured in HSS filters from the continuous cleanup of helium circulating through the reactor as well as FHS process gas. The applicant's evaluation of this release determined an offsite dose of 3.19 mrem (0.0319 mSv) TEDE. (The NRC staff's evaluation of postulated HSS releases is addressed in section 3.10.3.6 below.) Similarly, PSAR section 3.6.4 and PSAR table 3.2-5 address the DBA release of circulating activity within the HPB following an MD, including portions of the FHS that are not isolated. This event is bounded by a dose of 3.68 rem (0.0368 Sv) TEDE, which represents the release from a medium HPB break that is not isolated. These sources contain larger inventories of radionuclides because they represent the dust and radionuclide inventory collected and released from the entire HPB rather than the inventory present in the much smaller FHS, which contains a small fraction of the gas volume and fuel pebbles contained in the reactor HPB. Neither event credits holdup or filtration of the release. Therefore, the NRC staff determined that the applicant's preliminary conclusion of insignificant consequences (i.e., < 2.5 mrem (0.025 mSv) TEDE) from both the release from FHS filters and the release of FHS circulating activity is reasonable, and adequate safety margins are available without credit for SSCs to prevent releases or mitigate consequences.

3.10.1.4 Fuel in the Spent Fuel Storage System

The NRC staff reviewed PSAR section 3.10.4, which describes the evaluations of thermal challenges to fuel in the handling and storage phases, The applicant states that each SFSS canister is designed to store up to 7200 fuel pebbles. Each canister is filled in the CPF, and

filled canisters are transported on remotely operated vehicles through the inter-unit access tunnel to the SFISF.

Thermal Events Affecting Fuel in the Spent Fuel Canister Handling Phase

The NRC staff reviewed PSAR section 3.10.4.1.1, in which the applicant describes its assessment of thermal challenges from a loss of room cooling affecting a single canister in the handling phase. The applicant determined that the fuel temperatures have significant margins to fuel failure temperatures and that no oxidation is expected. The applicant indicates that this event may be screened out of the analyzed events in the FSAR because the event is slow in developing and there is sufficient time to provide an adequate response.

The NRC staff evaluated the thermal challenge to a single canister filled with recently discharged pebbles in the CPF prior to transfer into storage. The NRC staff noted that PSAR section 7.3.1.4 and PSAR figure 7.3.1-11, “Canister Processing Facility Plan View,” indicate that the CPF is a single story reinforced concrete structure with thick walls provided for shielding. Using dimensions from the figure, the NRC staff independently evaluated the expected response of canister temperature to a loss of room cooling, which included consideration of a tall canister cooled by natural convection and radiation to walls initially at a high environmental temperature of 40°C, and determined that fuel temperature would remain well below temperatures that could result in fuel damage or significant oxidation. Therefore, the NRC staff concluded that the preliminary SFSS design provides sufficient safety margins during normal operations and transient conditions by supporting adequate heat removal by passive means from loaded canisters.

Thermal Events Affecting Fuel in the Spent Fuel Canister Storage Phase

The NRC staff reviewed PSAR section 3.10.4.1.2, in which the applicant describes its assessment of thermal challenges from a loss of SFISF cooling affecting storage of canisters containing an inventory representative of 60 years of operation that bounds all phases of operation. Since the heat load is dominated by the most recently discharged fuel, the selected inventory includes a just-completed discharge of pebbles from one module.

PSAR section 7.3.1.3 describes the construction of the SFISF, the spent fuel storage racks (SFSRs), and SFCs. The SFISF is designed to accommodate approximately 660 SFCs on the SFSRs, such that each SFC receives natural circulation cooling air flow and protection from excessive challenges. The SFISF cooling air flow is provided through two vertical intake ducts along the North and South sides of the SFISF that draw air in above the maximum flood elevation and release it just above the interior floor. The SFSRs support the SFCs in an elevated position that allows cool air flow under and between the SFCs and prevents challenges to adequate cooling under accident conditions by preventing obstructions (e.g., pebble accumulation following a canister tip-over event) from developing. The heated air exits the SFISF through a central roof-mounted chimney that runs the length of the SFISF in an East-West direction. The geometry of stored pebbles established by the SFSRs and the SFCs prevents criticality. The applicant classifies the SFISF, the SFSRs, and the SFCs as NSRST for these functions. PSAR table 3.7-1 designates the SFISF as a seismic category IIA civil structure, and PSAR table 3.7-2 designates the racks and canisters as seismic category IIA to preserve dynamic characteristics and for seismic safety significance.

PSAR section 3.10.4.1.2 states that an evaluation of complete blockage of SFISF cooling ventilation flow for 7 days demonstrates that the temperature of fuel within the SFCs would retain significant margin to failure (i.e., temperature <300°C) and no oxidation would be expected. The applicant stated that the evaluation results show fuel temperature peaking around four days with heat transferred to surrounding structures and no credit for subsequent heat loss to the environment. PSAR section 1.1.4.3. 2 states that visual or remote monitoring of ventilation would be capable of detecting conditions that challenge residual heat removal. The applicant indicates that this event may be screened out of the analyzed events in the FSAR because the event is slow in developing and there is sufficient time to provide an adequate response.

The NRC staff evaluated the thermal challenge to SFCs stored in the SFISF, considering the potential for several SFCs to be loaded with recently discharged pebbles. The NRC staff reviewed the applicant's evaluation of complete blockage of ventilation inlet flow in audit. The NRC staff noted the importance of maintaining adequate cooling to the canisters with the greatest heat load because they are expected to reach significantly higher temperatures than other canisters in the passive cooling regime of the SFISF. Accordingly, the NRC staff focused its evaluation on the reliability of adequate cooling to higher heat load canisters in the preliminary design of the SFISF, SFSRs, and SFCs. The capacity of each SFISF provides assurance that the number of high heat load canisters (about 30 filled SFCs per module core) would always be a small fraction of that capacity, thereby assuring significant internal heat sink capacity would be available for those high heat load canisters following any challenge to external cooling. Furthermore, challenges to external cooling would be detectable through visual or remote monitoring, which supports prompt correction of the condition. The design of the SFSRs and SFCs minimizes the likelihood of obstructions or changes in fuel storage geometry that could adversely affect cooling under accident conditions, thereby assuring adequate natural circulation cooling of those high heat load canisters. Therefore, the NRC staff concluded that the preliminary design of the SFISF, SFSRs, and SFCs provides sufficient safety margins during normal operations and transient conditions by maintaining storage geometry and passive heat removal.

Mechanical Challenges Affecting Fuel in the SFSS

PSAR section 3.10.4.2 states that the applicant assumed no LBE would cause sufficient mechanical stress to challenge the fuel particle and fuel pebble radionuclide barriers and stated that fuel mechanical performance would be evaluated through testing and analysis to demonstrate the fuel response to selected events for the FSAR.

The NRC staff evaluated the potential for mechanical challenges to fuel pebbles in the SFSS. As noted in PSAR section 3.10.2.2, the minimum compressive strength specification for fuel pebbles is 17.5kN. The NRC staff confirmed that the value is consistent with specifications provided in the regulatory audit. The NRC staff performed simplified confirmatory calculations to evaluate the energy from short distance drops and maximum forces generated by collisions with unyielding surfaces. The NRC staff concluded that forces resulting from credible events affecting fuel pebbles within the SFSS, such as canister tip-over with subsequent pebble spillage, would be unlikely to challenge the integrity of the fuel pebble, although minor surface damage may occur affecting the fuel free zone. Therefore, the NRC staff determined that the applicant's assumption is reasonable, and no mechanical events are expected to challenge the fuel particle or fuel pebble radionuclide barriers.

3.10.1.5 Radwaste Treatment Building Sources

The NRC staff reviewed PSAR section 3.10.5, in which the applicant describes its assessment of postulated liquid and solid radioactive waste releases during normal and off-normal operations. The applicant calculated a 30-day dose at the EAB of 1.85 rem (0.0185 Sv) TEDE for a postulated release of the entire liquid and solid radionuclide inventory listed in PSAR tables 3.10-5 and 3.10-6, respectively. The applicant also evaluated a more realistic scenario that assumed a 1 percent airborne release fraction for in-process liquid waste (i.e., ion exchange media in vessels and low activity waste from floor and equipment drains) and unpackaged solid waste (i.e., unprocessed filters), which resulted in a dose of 7.88 mrem (0.0788 mSv) TEDE. The applicant determined that inclusion of these potential release sources in a future PRA would not lead to LBEs that exceed the F-C target and that no safety-significant SSCs would be needed to prevent or mitigate these releases.

The NRC staff evaluated the form and radionuclide content of the accumulated radioactive waste. The NRC staff finds the partial release credible, but the consequences are sufficiently low that the F-C target values would not be challenged. The NRC staff finds a release approaching the entire inventory would not be credible, except for unlikely events that affect a significant fraction of waste storage forms, such as a fire. The frequency of large fires is sufficiently low such that consequences of a postulated release of the entire liquid and solid radionuclide inventory would not challenge the F-C target values. Based on the foregoing, the NRC staff concluded that the preliminary liquid and solid radwaste system form and activity provide sufficient margin to the F-C target for challenges during normal operations and transient conditions and no safety-significant (i.e., NSRST) SSCs are necessary to prevent or mitigate postulated releases.

3.10.1.6 Helium Service System Sources Outside the Helium Pressure Boundary

The NRC staff reviewed PSAR section 3.10.6, in which the applicant describes its assessment of postulated releases of radionuclides collected within the HSS during normal and off-normal operations. The applicant calculated a 30-day dose at the EAB of 3.19 mrem (0.0319 mSv) TEDE for a postulated release of the entire HPS dust filter and low temperature adsorber radionuclide inventories listed in PSAR tables 3.10-7 and 3.10-8, respectively. The dust filters contain graphite dust contaminated with radionuclides and solid metals collected from the HPB, and the low temperature adsorbers capture gaseous contaminants, including radionuclides, from the HPB. The applicant states that the timing of the release for each component maximizes the radionuclide inventories.

The NRC staff evaluated the radionuclide content of the accumulated radionuclides collected by the HPS. The NRC staff finds that a release exceeding the F-C target would not be credible considering the preliminary maximum radionuclide inventories determined for the HPS. Based on the foregoing, the NRC staff concluded that the preliminary HPS design provides sufficient margin to the F-C target for challenges during normal operations and transient conditions and no safety-significant (i.e., NSRST) SSCs are necessary to prevent or mitigate postulated releases.

3.10.1.7 Neutron Start-Up Source

The NRC staff reviewed PSAR section 3.10.7, in which the applicant describes its assessment of postulated radiological impacts from a preliminary start-up source design containing 400 mCi

of unirradiated Cf-252 within a graphite shell the size of a standard fuel pebble. Through discussions in the regulatory audit, the NRC staff confirmed that the geometry of the source used in this analysis was an assumption based on early preliminary design and that the design at the time of the PSAR submittal was for a fixed source, as discussed in section 6.4.2 of this SE. LME did not perform a revised analysis on the radiological effects based on the limited impact it would have on the results of this analysis. The NRC staff agrees that this analysis is unlikely to impact the overall results of this analysis and finds it acceptable to support the CP.

The applicant's evaluation addressed the 30-day dose at the EAB from the alpha-decay of Cf-252 without dispersal or aerosolization of the source, the integrated effects of the unshielded neutron field generated by spontaneous fission of the Cf-252 source, and the gamma field resulting from decay of spontaneous fission products. The PSAR description notes that none of the alpha particles generated by Cf-252 decay contribute to inhalation or immersion dose because the source is encapsulated and that the neutrons are instantaneously released with no credit for shielding. The applicant determined that a conservative dose would be 15.75 mrem (0.1575 mSv) TEDE with the source 300 meters from the EAB, and that dose results almost entirely from the unshielded neutron field.

The NRC staff evaluated the radiological effects from the neutron start-up source. The NRC staff finds the effects of an encapsulated source that can be easily shielded to be negligible. Therefore, the NRC staff determined that the neutron source with intact encapsulation provides sufficient margin to the F-C target for challenges during normal operations and transient conditions and no safety-significant (i.e., NSRST) SSCs are necessary to prevent or mitigate radiological impacts from the encapsulated source. The NRC staff will evaluate the safety significance of the encapsulation at the OL application stage when more information is available about the final source type and installation.

3.10.1.8 Results

The PSAR indicated that the preliminary conclusions, as discussed in the prior sections of this SE, will ultimately be supported by incorporating all radiological sources to support the final design in the FSAR. The NRC staff notes the importance of the SR fuel pebble and its resistance to damage from mechanical or thermal events in preserving its RSF of retaining radionuclides. PSAR section 3.10.2.2 describes the fuel pebble mechanical strength, and PSAR section 6.4.2 describes the fuel design and qualification. In addition, general experiments with graphite coated fuel in oxidizing environments demonstrate the fuel capability to retain radionuclides to moderately elevated temperatures in air. Therefore, the NRC staff's detailed review of the results focused on postulated events affecting more than one fuel pebble, such as the representative FHS mechanical event involving HPB failure and the representative SFSS thermal challenge, because these events are expected to produce bounding radiological consequences due to the potential to challenge fission product barriers in recently irradiated fuel.

The NRC staff evaluated the assessment of fuel damage and resulting consequences from a representative mechanical challenge in the FHS. The ejection of pebbles under pressure provides the force necessary to mechanically damage barriers engineered in the pebble design. The applicant's evaluation of a loss of conveyance gas flow and a complete break in the FHS conveyance piping demonstrate that the FHS NSRST design functions described in PSAR section 7.3.6 to minimize challenges to fuel and minimize the likelihood of HPB failures are adequate to provide appropriate safety margin and address uncertainty.

The NRC staff also evaluated the consequences of a representative thermal challenge to fuel pebbles stored in the SFSS. The relocation of pebbles to geometries that degrade cooling or create the potential for critical heat generation could challenge the fuel barriers over long periods through oxidation. The applicant's evaluation of degraded cooling to fuel stored within the SFISF demonstrated that the fuel barriers would be maintained with the fuel in a uniform storage geometry that provided airflow around each fuel canister and prevented criticality. PSAR section 7.3.1.3 describes NSRST functions of the SFISF structure, the SFSS racks, and the SFSS fuel canisters to maintain fuel storage geometry, and PSAR section 3.9 demonstrates the margin to criticality the SFSS fuel storage geometry provides. These NSRST functions provide an appropriate safety margin for fuel in the SFSS storage by maintaining the assumed storage geometry under challenging design conditions described in PSAR section 7.3.1.3 and, thereby, reasonably address uncertainty.

The above evaluations of non-core radionuclide sources at the preliminary design stage reasonably demonstrate that no postulated events affecting these sources would exceed the F-C target. The SR fuel pebble design described in PSAR section 6.4.2 provides the necessary radionuclide retention functions for all postulated events involving fuel. The NRC staff will evaluate the results of the PRA-supported analyses in the FSAR to confirm these results at the OL application stage.

3.10.2 Conclusion

Based on its review, the NRC staff finds that the evaluation of non-core related source radiological analyses, described in PSAR section 3.10, is consistent with the applicable regulatory guidance in RG 1.253 and therefore meets the following regulatory requirements:

- 10 CFR 50.34(a)(1)(ii)(D) with respect to describing the safety features engineered into the facility and those barriers that must be breached before a release of non-core radioactive material could reach the environment; and

- 10 CFR 50.34(a)(4) with respect to providing a preliminary analysis of non-core radioactive material sources and associated SSCs with the objective of assessing risk to public health and determining margins of safety.

3.11 Supplemental Evaluations of Plant Operating States

PSAR section 3.11 describes the qualitative evaluations performed to determine whether the DBAs identified in section 3.6 would be more limiting if they were to occur during operating modes other than full-power operations. PSAR table 3.11-1 defines the LMGS module operating modes.

The applicable regulatory requirements for the evaluation of the non-full-power POSs are as follows:

- 10 CFR 50.34(a)(1); and

- 10 CFR 50.34(a)(4).

The applicable guidance for the evaluation of the non-full-power POSs are as follows:

- RG 1.233, Revision 0, which endorses NEI 18-04, Revision 1, with clarifications;
- RG 1.247, which endorses ASME/ANS RA-S-1.4-2021 with exceptions; and
- RG 1.253, which endorses NEI 21-07, Revision 1, with clarifications and additions.

3.11.1 Technical Evaluation

PSAR section 3.11 states that qualitative analyses were performed to assess the DBAs evaluated in PSAR section 3.6 at non-full-power conditions. The NRC staff evaluated the information in PSAR section 3.11 by reviewing the approach to performing supplemental evaluations and the results of the supplemental evaluations.

PSAR section 3.11.1 states: "The criterion used to identify the limiting events is the offsite dose figure of merit...[t]he comparison of dose from the full-power DBA to a proposed DBA in other plant modes is considered based in part on maximizing fuel temperature." The NRC staff finds that the use of the radiological dose as the figure of merit is reasonable because it provides the consequence value for comparison against the F-C target. Additionally, the NRC staff finds that the use of fuel temperature is a reasonable surrogate for dose because it correlates with radiological consequences due to (1) most radionuclides residing within the fuel, and (2) radiological release from the fuel being a diffusion-dominant process (i.e., driven by temperature).

The NRC staff's review finds that the applicant's extension of the safety analysis to states other than full-power initiating conditions using qualitative analysis acceptable because criterion A.3.3 from RG 1.253 identifies the acceptable PRA scope for a CP application as "internal events hazards group for the reactor in the at power [plant operating state]."

Based on the above, the NRC staff finds the approach to performing supplemental evaluations for preliminary analysis to be acceptable.

Results of Supplemental Evaluations

PSAR sections 3.11.2 through 3.11.11 provide the results of the qualitative supplemental evaluations for the DBAs analyzed in PSAR section 3.6. The NRC staff's summary and evaluation of the supplemental evaluation results, based on the information provided in PSAR section 3.11, is provided in table 3.11-1 of the SE, below.

Table 3.11-1: NRC Staff Summary and Evaluation of Supplemental DBA Evaluations

DBA Identifier	PSAR Section	Supplemental Evaluation Summary	NRC Staff Evaluation
Control Rod Withdrawal	3.6.1	Positive reactivity insertion could be more limiting at lower power and in Modes 2-3 since the control rods could	The evaluation of control rod withdrawal at lower powers should be performed because: (1) control rod insertion affects the amount and rate of reactivity

DBA Identifier	PSAR Section	Supplemental Evaluation Summary	NRC Staff Evaluation
		be inserted farther. These cases will be explicitly evaluated for impact of reactivity insertion in the FSAR.	inserted, and (2) initiating the event from lower power levels and structure temperatures changes the plant response such that different RT signals (e.g., power-rate) may be required. As described in RG 1.253, the evaluation of control rod withdrawal DBA at power levels other than full power is not required during the CP stage.
Loss of Primary Flow	3.6.2	The full-power operating core temperatures and subsequent fuel temperatures from decay heat following shutdown exceed what would be expected starting from a lower power operating mode (including Mode 2), and for that reason, the full-power case is more limiting.	The NRC staff finds it reasonable that the full-power case would be limiting because decay heat power levels are higher. However, additional analyses may be warranted in the FSAR if the heat removal capacity of the RCCS or the performance of other SR SSCs are affected by the power level. As described in RG 1.253, the evaluation of heat removal capacity of RCCS or performance of other SR SSCs at power levels other than full power is not required during the CP stage.
Small HPB Depressurization	3.6.3	The full-power case bounds the lower power cases and Modes 2 and 3 because of the higher core temperatures and decay heat at full power.	The NRC staff finds it reasonable that the full-power case would be limiting because decay heat power levels are higher. However, additional analyses at other power levels may be warranted in the FSAR for the DBA using detailed design information on startup and non-equilibrium core to show that the full-power SD case remains bounding. As described in RG 1.253, the evaluation of the SD DBA at power levels other than full power is not required at the CP stage.

DBA Identifier	PSAR Section	Supplemental Evaluation Summary	NRC Staff Evaluation
Medium HPB Depressurization	3.6.4	Same as SD	Same as SD
Loss of Steam Generator Feedwater	3.6.5	The full-power operating core temperatures and subsequent fuel temperatures from decay heat following a shutdown exceeds what would be expected starting from a lower power operating mode (including Mode 2), and for that reason the full-power case is the most limiting.	The NRC staff finds it reasonable that the full-power case would be limiting because decay heat power levels are higher. However, additional analyses at other power levels may be warranted in the FSAR for the DBA using detailed design information on startup and non-equilibrium core to show that the full-power Loss of Steam Generator case remains bounding. As described in RG 1.253, the evaluation of the Loss of Steam Generator Feedwater DBA at power levels other than full power is not required at the CP stage.
Main Feedline Break	3.6.6	Same as Loss of Steam Generator Feedwater	Same as Loss of Steam Generator Feedwater
Main Steam Line Break	3.6.7	It is possible that the startup core would behave somewhat differently due to the presence of more moderator and a different fuel mixture. For that reason, the steam line break LBE will be more fully evaluated for the FSAR.	The NRC staff agrees that the evaluation of MSLB should be evaluated at multiple power levels and core configurations (i.e., startup vs equilibrium) because: (1) the response of the reactor core is expected to be different, and (2) the performance of SR SSCs may be affected by power level in the final design. As described in RG 1.253, the evaluation of the MSLB DBA at power levels other than full power is not required at the CP stage.
Steam Generator Tube Leak	3.6.8	The full-power event is bounded by the steam generator tube rupture DBA. Steam generator tube leaks at lower power levels (including Mode 2) are also expected to be bounded by rupture events	The NRC staff agrees that the steam generator tube leak would be bounded by the steam generator tube rupture because: (1) a rupture is a more severe leak, and (2) the same SR SSCs would be credited to mitigate both events.

DBA Identifier	PSAR Section	Supplemental Evaluation Summary	NRC Staff Evaluation
		beginning at the same power levels.	
Steam Generator Tube Rupture	3.6.9	The startup core may yield more limiting results than the equilibrium core as it comprises a different fuel mixture. For these reasons, a more complete analysis will be evaluated at the FSAR stage.	The NRC staff agrees that the evaluation of the main steam line rupture should be evaluated at multiple power levels and core configurations (i.e., startup vs equilibrium) because: (1) the response of the reactor core is expected to be different, and (2) the performance of SR SSCs may be affected by power level in the final design. As described in RG 1.253, the evaluation of Steam Generator Tube Rupture DBA at power levels other than full power is not required at the CP stage.
Seismic	3.6.12	Pebble compaction may yield higher reactivity insertion in Mode 2 and for the startup and non-equilibrium cores. An evaluation plan has been developed and will be completed at the FSAR stage.	The NRC staff expects the seismic event be evaluated, or suitably justified, over a range of power levels and core configurations because: (1) the response of the reactor core is expected to be different, and (2) the performance of SR SSCs may be affected by power level in the final design. As described in RG 1.253, the evaluation of the Seismic DBA at power levels other than full power is not required at the CP stage.

Based on the information provided in PSAR Section 3.11, the NRC staff finds the results of the supplemental DBA evaluations to be acceptable because:

- the applicant provided qualitative justification for the DBAs where at power cases bound the non-equilibrium modes; and
- provided a commitment to perform an evaluation at the OL application stage for the cases where events starting from the non-equilibrium mode could be more limiting.

3.11.2 Conclusion

Based on the above, the NRC staff finds that the approach and results of the supplemental DBA evaluations are acceptable. Therefore, the NRC staff finds that the analyses of the non-full-power POSs described in PSAR section 3.11 are consistent with the applicable regulatory guidance and meet the regulatory requirements identified in 10 CFR 50.34 for the issuance of a CP.

3.12 Aircraft Impact Assessment

The NRC staff reviewed Part VII of the CP application which includes a description of the LMGS aircraft impact assessment (AIA).

The applicable regulatory requirements for the submission of a description of the LMGS AIA at the CP stage are as follows:

- 10 CFR 50.150(a);
- 10 CFR 50.150(b); and
- 10 CFR 50.34(a)(13), (requiring, by cross reference, that the PSAR include the description of the AIA design features and functional capabilities).

Paragraph 50.150(b) of 10 CFR requires that a CP application's PSAR must include a description of: (1) the design features and functional capabilities identified in 10 CFR 50.150(a)(1) and (2) how the design features and functional capabilities identified in 10 CFR 50.150(a)(1) meet the assessment requirements in 10 CFR 50.150(a)(1) (requiring that each applicant CP perform a realistic design-specific assessment of the effects on the facility of the impact of a large commercial aircraft) to show that, with reduced use of operator actions: (i) the reactor core remains cooled, or the containment remains intact; and (ii) spent fuel cooling or spent fuel pool integrity is maintained. All that is required in a CP application is "a description" of the design features and functional capabilities that will meet the requirements of 10 CFR 50.150(a)(1). An applicant is not required to submit the AIA (as opposed to a description) to the NRC in its application.

Review of the AIA is not part of the reviews performed on LME's CP application. Instead, consistent with the above regulations, the NRC staff reviewed the adequacy of the description of the design features and functional capabilities contained in the CP application.

The applicable guidance document for evaluation of the AIA is as follows:

- RG 1.217, Revision 0, "Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts" (ML092900004), which endorses the guidance in NEI 07-13, Revision 8, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs."

3.12.1 Technical Evaluation

Because the LMGS design uses functional containment and does not include the spent fuel pool typical of LWRs, the CPA identifies the relevant AIA acceptance criteria in 10 CFR 50.150(a)(1)

that are applicable to the LMGS design. Specifically, LME identified the design criteria for LMGS as:

(i) The reactor core remains cooled; and

(ii) Spent fuel cooling is maintained.

The NRC staff determined that these criteria are the appropriate 10 CFR 50.150(a)(1) criteria to be applied to the LMGS design, given its design features described above.

The NRC staff's review focused on the adequacy of the description of the design features and functional capabilities credited by LME to demonstrate that the reactor core remains cooled and spent fuel cooling is maintained. The application described how the design features and functional capabilities include the SST and SFISF which provide overall protection of SSCs housed within these structures, including SSCs that support reactor core and spent fuel cooling, against adverse effects of a large commercial aircraft impact in accordance with 10 CFR 50.150(a)(2). This aircraft type is representative of long-distance flights within the United States and includes a typical aviation fuel load.

The NRC staff reviewed LME's implementation of the guidance in NEI 07-13, as endorsed by RG 1.217. The NRC staff's review of LME's AIA description identified that NEI 07-13 is used to assess the potential for perforation of the structures within the scope of the AIA, to define material properties and failure criteria for the assessment of the global response of the structures, to characterize loading parameters, and to inform the development and implementation of analysis assumptions and overall parameters for both local damage and global response assessments. The NRC staff determined that LME's description of the assessment methodology in these areas is consistent with RG 1.217.

The NRC staff reviewed the description of design features and functional capabilities relied upon by LME for ensuring that the reactor core remains cooled, which include the SST walls and roof, the reactor support structures (RXSTs), and passive heat removal via the RCCS. In addition to the SST (which houses and protects the RXSTs and RCCS), LME identified the RXSTs and RCCS as key features relied upon to ensure that the reactor remains cooled and stated that these key features meet RSFs 1.4 and 1.2, and related sub functions. Specifically, by ensuring that the geometry of RXSTs is maintained, LME ensures that the reactor modules and related RCCS continue to be adequately supported by the RXSTs, thereby maintaining the control rod insertion and RCCS passive heat removal capabilities unaffected. The analyses performed in support of AIA indicate that RXST geometry is not adversely affected by either local or global damage mechanisms experienced by the SST. The evaluation of local damage demonstrated that an aircraft impact would not result in perforation or scabbing damage inside the SST, and a conservative global evaluation showed that the global response of the SST does not cause adverse physical interaction with the RXSTs.

The NRC staff reviewed the description of design features and functional capabilities relied upon by LME for ensuring that spent fuel cooling is maintained, which include the SFISF walls and roof, the canister support structures, the SFCs, and passive heat removal. LME describes that these features are designed hierarchically to maintain spent fuel cooling. Specifically, LME states that the SFISF building structure will be relied upon to prevent perforation, thereby preventing adverse physical interaction with fuel support structures, SFCs, and their respective

passive heat removal capabilities such that spent fuel cooling is maintained without reliance on active components or operator actions.

The NRC staff reviewed LME's description of the approach for protecting building ventilation openings and equipment doors, which includes a systematic evaluation of all building openings. Based on this evaluation, and consistent with NEI 07-13, LME identifies large building openings and establishes corresponding protection requirements. These protection requirements set performance objectives for protective features at large openings that align with the building's performance objectives - including the capability to preclude missile perforation - and require use of the codes and standards applicable to the building's design. The NRC staff finds LME's methodology for protecting building ventilation openings and equipment doors reasonable because LME will design and qualify protection for each large opening to achieve the same performance as the building envelope, specifically, to preclude missile perforation, in accordance with NEI 07-13/RG 1.217 methods and the applicable codes and standards.

3.12.2 Conclusion

Based on the evaluations described above, the NRC staff finds that the preliminary LMGS AIA description provided in Part VII of the CP application is sufficient for the CP stage, is consistent with the applicable guidance, and meets the regulatory requirements of 10 CFR 50.150(a), 10 CFR 50.150(b), and 10 CFR 50.34(a)(13). The NRC staff will review the final AIA information during the review of a future OL application.

3.13 Mitigation of Beyond Design Basis Events

The NRC staff reviewed PSAR section 3.13, which states: "LME is not requesting design finality [for mitigation of BDBEs] in the CPA. Compliance with 10 CFR 50.155(b) through (d) will be provided with the [OL application]." The NRC staff notes that RG 1.253, Section C, Subsection 4, "Licensing Basis Events," (ML23269A222) states that: "[a]pplicants for a CP that are not requesting design finality for mitigation of specific beyond-design-basis events reflected in 10 CFR 50.155(a) and applicants for a DC are not required to provide information on this topic." Therefore, the NRC staff finds that information regarding the mitigation of BDBE is not required at the CP application stage.

4 INTEGRATED EVALUATION

This chapter of the safety evaluation (SE) describes the U.S. Nuclear Regulatory Commission (NRC or Commission) staff's review and evaluation of the Long Mott Generating Station's (LMGS) preliminary safety analysis report (PSAR) chapter 4, "Integrated Evaluation." The integrated evaluations described in PSAR chapter 4 consider the impact from anticipated operational occurrences (AOOs), design-basis events (DBEs), and beyond design bases events (BDBEs) on the overall plant risk performance and the adequacy of defense-in-depth (DID).

4.1 Overall Plant Risk Performance Summary

PSAR section 4.1 summarizes the overall plant risk to public health and safety from plant operation and compares analysis results to the risk metrics described in Nuclear Energy Institute (NEI) 18-04, Revision 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Revision 1 (ML19241A472). The applicable regulatory requirements for the evaluation of PSAR section 4.1 are as follows:

- 10 CFR 50.34(a)(4).

The applicable guidance documents for the evaluation of overall plant risk performance are as follows:

- Regulatory Guide (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 NEI 18-04, 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 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), with clarifications and additions.

4.1.1 Technical Evaluation

The NRC staff reviewed PSAR section 4.1 against the guidance discussed above to assess whether the overall integrated plant performance meets the established risk metrics in RG 1.253, Section C.5, "Integrated Evaluations." The probabilistic risk assessment (PRA) used to assess the overall integrated plant risk performance is evaluated in section 3.1 of this SE. The preliminary assessment of the overall integrated plant performance is based on the scope of the PRA, and a comparison is made against the three cumulative plant risk performance metrics from RG 1.253 and is evaluated below.

The NRC staff reviewed PSAR section 4.1.1, which states that the LMGS estimated total mean frequency of exceeding a site boundary dose of 100 mrem (1 mSv) is 3.49E-04/plant-year. The NRC staff compared this against the RG 1.253 criteria that the total mean frequency of

exceeding a site boundary dose of 100 mrem (1 mSv) from all non-design-basis accidents (DBA) licensing basis events (LBEs) should not exceed 1/plant-year.

PSAR section 4.1.2 states that the LMGS estimated total mean average individual risk of early fatality within 1 mile (1.61 km) of the exclusion area boundary (EAB) is 0/plant-year. The NRC staff compared this against the RG 1.253 criteria that the average individual risk of an early fatality within 1 mile (1.61 km) of the EAB from all non-DBA LBEs should not exceed 5E-7/plant-year.

PSAR section 4.1.3 states that the LMGS estimated average individual total mean risk of latent cancer fatalities within 10 miles (16.1 km) of the EAB is 7.15E-08/plant-year. The NRC staff compared this against the RG 1.253 criteria that the average individual risk of latent cancer fatalities within 10 miles (16.1 km) of the EAB from all non-DBA LBEs should not exceed 2E-6/plant-year.

These three cumulative plant risk performance metrics are consistent with RG 1.233. NEI 18-04, which is endorsed by RG 1.233 with clarifications, describes the three cumulative risk metrics that need to be met to evaluate the integrated plant performance in any application based on the licensing modernization project (LMP) methodology. As described in NEI 18-04, an applicant using the LMP evaluates the integrated risk of all LBEs against three cumulative risk metrics. The first cumulative risk metric is based on the annual radiation dose limit for individual members of the public as specified in Title 10 of the *Code of Federal Regulations* (10 CFR) Paragraph 20.1301(a)(1). The other two cumulative risk metrics ensure that the NRC safety goal quantitative health objectives (QHOs)[1] are satisfied. These three cumulative risk metrics as defined in NEI 18-04 are the same as those stated above for RG 1.253.

To further understand the summarized results of the preliminary assessment in PSAR section 4.1, the NRC staff audited Document ID number 002200, “Probabilistic Risk Assessment: Risk Integration – Preliminary Scoping,” Revision 5. The applicant’s preliminary assessment demonstrated that the LMGS preliminary design satisfies the three risk metrics with margin based on the preliminary PRA at the construction permit (CP) application stage. The radiological consequence methodology used to calculate these cumulative risk metrics is described in PSAR section 3.1.5.1 and evaluated by the NRC staff in section 3.1.5 of this SE. The risk integration analysis is evaluated by the NRC staff in section 3.1.1. Consistent with RG 1.253, the applicant assesses the items outside of the CP PRA scope using other approaches (e.g., risk-informed supplemental evaluations and the use of design-basis hazard levels (DBHLs)) to support the CP application. These items are not included in the applicant’s assessment of the overall plant risk in PSAR section 4.1. PSAR sections 3.10 and 3.11 include the results of the applicant’s supplemental evaluations and PSAR section 6.1.1 includes the use of DBHLs. These items are evaluated in sections 3.10, 3.11, and 6.1 of this SE. Although limited in scope, the applicant’s approach is reasonable given the preliminary stage of the design and PRA, that the estimated overall plant risk meets the metrics with margin, and the applicant plans to finalize the overall plant risk performance assessment at the operating license (OL) stage.

Therefore, the NRC staff finds that the applicant’s preliminary assessment of the overall integrated plant risk performance of the LMGS is reasonable and consistent with the applicable guidance in RG 1.233 and RG 1.253 at the CP stage. The NRC staff will confirm that the final

[1] “Safety Goals for Operations of Nuclear Power Plants; Policy Statement; Republication,” *Federal Register*, Vol. 51, No. 162, August 21, 1986, pp. 30028-30033 (51 FR 30028).

design meets the cumulative plant risk performance metrics during the evaluation of the final safety analysis report (FSAR) as part of the review of a future OL application.

4.1.2 Conclusion

Based on its review, the NRC staff finds that the overall plant risk performance, as described in PSAR section 4.1, is consistent with the applicable regulatory guidance and meets the regulatory requirements in 10 CFR 50.34(a)(4).

4.2 Defense-in-Depth

PSAR section 4.2 describes the applicant's approach to evaluating DID adequacy. The PSAR states that the approach is based on the guidance described in Section 5 of NEI 18-04 and the Chapter 4 structure and associated guidance provided in NEI 21-07, and includes plant capability DID, programmatic DID, and an integrated evaluation of DID.

The applicable regulatory requirements for the evaluation of DID are as follows:

- 10 CFR 50.34(a)(1) and (a)(4); and
- 10 CFR 50.35, "Issuance of construction permits."

The applicable guidance documents for the evaluation of DID adequacy are as follows:

- RG 1.233, Revision 0, which endorses NEI 18-04, Revision 1, with clarifications.
- RG 1.253, Revision 0, which endorses NEI 21-07, Revision 1, with clarifications and additions.

4.2.1 Technical Evaluation

PSAR section 4.2 states that: "[t]he systematic approach to evaluating DID adequacy is based on the guidance described in Section 5 of NEI 18-04 and the Chapter 4 structure and associated guidance provided in NEI 21-07." The NRC staff's review of the PSAR identified that the scope of the DID evaluation described in the PSAR is limited to the evaluation of plant capability DID, which the NRC staff finds to be consistent with the discussion in NEI 21-07, Section 4.2 for the content-of-application for a CP.

PSAR table 4.2-1 provides a summary of the plant capability DID attributes from NEI 18-04, the focus of the DID adequacy evaluation as discussed in NEI 18-04, and Long Mott Energy's (LME's) implementation of the evaluation for the LMGS. The PSAR indicates that detailed documentation of the DID adequacy evaluation is maintained in plant records and summarized in the PSAR.

For the plant capability DID evaluation, the NRC staff evaluated the information in PSAR section 4.2 against the criteria in NEI 21-07 for LBE margin, layers of defense, single feature reliance, and prevention-mitigation balance in sections 4.2.1.1 through 4.2.1.4 of this SE, respectively. In section 4.2.1.5 of this SE, the NRC staff additionally evaluated the applicant's

implementation of the integrated decision-making process (IDP) and IDP panel (IDPP), which support the applicant's DID adequacy evaluation as well as other topics such as safety classification and identification of special treatments.

4.2.1.1 LBE Margin

The NRC staff evaluated the information in PSAR section 4.2 against the following criterion from Section C.4.2.1.1 of NEI 21-07, as endorsed by RG 1.253, to ensure sufficient LBE margins in the LMGS design:

- Risk margins for individual risk-significant LBEs should be provided using both mean and 95th percentile values.

The NRC staff evaluated the information in PSAR section 4.2.1.1, and associated PSAR tables 4.2-2 and 4.2.3 to: (1) ensure that the tables include all risk-significant LBEs and (2) ensure that the information in the tables show margin to the frequency-consequence (F-C) curve. Specifically:

- The NRC staff compared the list of LBEs identified in PSAR tables 4.2-2 and 4.2-3 to the list of risk-significant LBEs identified by the NRC staff and provided in table 3.2-1, "Risk-Significant AOOs, DBEs, and BDBEs," in this SE and finds that the tables include all risk-significant LBEs.

- The NRC staff examined the mean frequency margin and mean dose margin in PSAR table 4.2-2, and the 95th frequency margin and 95th dose margin in PSAR table 4.2-3 and confirmed that all the values show margin to the F-C curve.

Based on this information, the NRC staff finds that the applicant meets the acceptance criterion for LBE margin from NEI 21-07 as endorsed by RG 1.253 because the PSAR provides information that shows that all risk-significant LBEs have margin to the F-C curve.

4.2.1.2 Layers of Defense Evaluation

The NRC staff evaluated the information in PSAR section 4.2 against the following criteria from Section C.4.2.1.2 of NEI 21-07, as endorsed by RG 1.253, to ensure that sufficient layers of defense are present in the LMGS design:

- The applicant should provide a summary identification of the layers of defense for each risk-significant LBE and describe the extent of independence between different LBE layers of defense.

- The LBE sequences should include the contributions from common cause failures in accordance with the PRA standard.

- The applicant should identify any risk-significant common cause failures that were not eliminated by design along with identification of the layers of defense impacted (i.e., layer independence impacts or reactor independence impacts) for risk-significant LBEs and the programmatic special treatments applied.

- The applicant should include an affirmative statement that there is no overreliance on a single layer of defense for any risk-significant LBE.

- The applicant should provide separate discussions of programmatic actions added as a result of a significant dependence on any single layer attribute evaluation. The description should be sufficient to identify the DID objectives requiring those actions.

As discussed in Section C.4.2.1.2 of NEI 21-07, the layers of defense evaluation should be based on the quantitative and qualitative guidelines provided in Table 5-2 of NEI 18-04.

Layers of Defense Evaluations

The NRC staff reviewed PSAR section 4.2.1.2, which describes five layers of defense. Layer 1 acts to prevent off-normal operation and AOOs; layer 2 controls abnormal operation, detects failures, and prevents DBEs; layer 3 controls DBEs within analyzed design-basis conditions and prevents risk-significant BDBEs; and layers 4 and 5 control severe plant conditions, mitigate consequences of BDBEs, deploy adequate offsite protective actions, and prevent adverse impacts on public health and safety. The NRC staff reviewed the description of these five layers and finds them to be consistent with Table 5-2 of NEI 18-04.

The NRC staff reviewed PSAR tables 4.2-4 through 4.2-6, which provide evaluations of layers 2, 3, and 4 and 5, respectively, against the quantitative and qualitative criteria from Table 5-2 of NEI 18-04. The NRC staff reviewed the format and structure of PSAR tables 4.2-4 through 4.2-6 and finds them to be consistent with section 4.2.1.2, “Layers of Defense Evaluation,” of NEI 21-07 as endorsed by RG 1.253. The NRC staff also confirmed that these PSAR tables contain the appropriate risk-significant LBEs, as listed in table 3.2-1 of this SE. The applicant also stated that while the content of PSAR section 4.2 is limited to risk-significant LBEs, consistent with the content-of-application guidance provided in NEI 21-07, the full scope of LBEs were considered in the DID evaluations used to identify non-safety-related with special treatment (NSRST) PRA safety functions (PSFs) necessary for adequate DID. The NRC staff finds this to be consistent with the DID evaluation process as discussed in NEI 18-04, which indicates that defense layer qualitative and quantitative guidelines are to be applied to all LBEs.

For layer 1, the applicant identified that provisions that act to prevent AOOs are non-safety-related with no special treatment (NST), and any structures, systems, and components (SSCs) that are needed to reduce the frequency or severity of or prevent initiating events and are found to be necessary to provide adequate DID (and are thus classified as safety-related (SR) or NSRST) are identified in the evaluation of other layers of defense. The NRC staff finds this approach for the layer 1 evaluation acceptable because it is consistent with the discussion in NEI 21-07, Section C.4.2.1.2.

For layer 2, PSAR table 4.2-4 provides the layers of defense evaluation for risk-significant AOOs. The applicant interpreted the guidance for layer 2 in Table 5-2 of NEI 18-04 as an objective for AOOs to rely on NST and NSRST SSCs for mitigation to minimize challenges to SR SSCs. The center column of PSAR table 4.2-4 provides a list of PSFs that serve to mitigate the AOO, and the right column provides the PSFs that prevent challenges to the required safety functions (RSFs). The applicant minimized challenges to the RSFs by considering the purpose of each RSF and identifying at least one PSF that provides DID for that purpose and elevating it to NSRST for DID. Where more than one PSF provided DID, some PSFs that prevent

challenges to RSFs remained NST based on adequate DID already being achieved. Additionally, some AOOs in PSAR table 4.2-4 rely on an RSF for mitigation (e.g., SPLPF-08, which relies on RSF 1.1.2, and SSGTL-03, which relies on RSF 1.2.1). In these cases, there is a PSF that prevents challenges to the RSF. Due to conservative assumptions for failure frequencies, the applicant identified that the LBEs with these functions failed to appear in the AOO range for the CP. The applicant considered this to be reasonable for adequate DID because the qualitative objective in NEI 18-04, Table 5-2 is to minimize the frequency of challenges to the RSFs. The NRC staff developed a summary of the RSFs that are protected by NSRST PSFs and the associated NSRST SSCs in table 5.5-1 of this SE. The NRC staff finds the applicant's approach to layer 2 to be acceptable for a CP since it reflects a preliminary DID evaluation and the applicant identified PSFs responsible for reducing the frequency of challenges to RSFs. The NRC staff will review the applicant's justification for DID adequacy on an initiating event family basis, demonstrating that there is DID for each event type for the final evaluation at the OL application stage.

For layer 3, PSAR table 4.2-5 provides the layers of defense evaluation for risk-significant DBEs. The applicant interpreted the layer 3 considerations to apply to DBEs relying on performance of RSFs for mitigation and BDBEs relying on RSFs for prevention. The middle column of PSAR table 4.2-5 provides the PSFs and RSFs that mitigate the DBE. The NRC staff notes that several DBEs rely on PSF 1.2.2.1.3, "Maintain active heat removal during AOOs"; as discussed in section 5.2.1 of this SE, the name implying a connection to AOOs is an artifact of the applicant's process in determining PSFs, and PSF 1.2.2.1.3 is applicable to any type of transient.

The qualitative guideline from NEI 18-04, Table 5-2 is for no single design or operational feature to be relied upon to meet the quantitative objective for all DBEs. The applicant interpreted this as relating to all DBEs, rather than DBEs individually. For the current scope of events considered in the PRA, this approach is appropriate. However, the NRC staff observes that in future analyses anticipated for a future OL application, it may not be appropriate to consider this criterion across all DBEs because there will likely be DBEs evaluated with sources of radionuclides outside of the core; these DBEs would rely on different safety functions, which if aggregated with DBEs involving the core could give a false determination on DID adequacy. The applicant identified that no single design or operational feature was relied upon because DBEs have functions available to maintain subcriticality, maintain the integrity of the helium pressure boundary (HPB), or maintain active heat removal.

For DBEs, because multiple functions are available to bring the reactor subcritical and maintain active heat removal to prevent challenges to fuel temperature limits, and additional radionuclide retention is provided by the HPB if temperature limits are exceeded, the NRC staff finds that there was no undue reliance on a single design or operational feature for DBEs for the CP. The NRC staff finds the applicant's approach to layer 3 to be acceptable for a CP. The NRC staff will review the applicant's evaluation results demonstrating that for DBEs there is no overreliance on a single design or operational feature on an initiating event family basis for the final evaluation at the OL application stage.

For layers 4 and 5, PSAR table 4.2-6 provides the layer of defense evaluation of the risk-significant BDBEs. The middle column of PSAR table 4.2-6 summarizes the credited PSFs and RSFs for the BDBEs. As with the layer 3 qualitative guideline, the applicant evaluated whether a single design or operational feature is relied upon across all BDBEs to maintain individual risks within the QHOs. PSAR section 4.2.1.2 states that a single design or operational feature is not

relied on because the fuel provides an SR means to retain radionuclides and the HPB provides an independent layer of protection against radionuclide release that does not experience the same failure modes as the fuel. The reliability of PSF 2.7, “Maintain the HPB pressure integrity,” prevents high frequencies for depressurization transients, which rely on the reactor cavity cooling system for heat removal to maintain fuel temperatures low enough to limit radionuclide releases. The applicant indicates that these functions together are able to maintain risks below the QHOs. Because the applicant met the quantitative objective of ensuring individual risks from LBEs are below the QHOs, and the qualitative objective of ensuring no single barrier or feature is relied upon to limit releases, as discussed above, the NRC staff determined the applicant’s approach to layers 4 and 5 is acceptable for a CP. As noted above for the layers 2 and 3 evaluations, the NRC staff will review DID adequacy and single feature reliance on an initiating event family basis for the final evaluation at the OL application stage.

Independence between defense layers

PSAR section 4.2.1.2 provides a discussion on independence between layers. The applicant states that layer 2 fulfils some functions also provided by RSFs, such as controlling heat generation, controlling heat removal, and controlling water/steam ingress, in a manner that is functionally independent from the RSFs in layer 3. As discussed above, these functions help to maintain the integrity of the fuel and thus limit radionuclide releases for AOOs. Layers 4 and 5 provide sufficient independent means to limit the fraction of source term released from the HPB as discussed above. As such, the NRC staff considered the applicant’s approach to defense layer independence to be acceptable for the CP stage.

Common cause failures

Regarding the criteria related to the evaluation of common cause failures (CCFs), PSAR section 4.2.1.4 states that: “[f]or each SSC, likelihood of RSF/PSF failure is minimized by considering potential for common cause failures, single failure criterion, fail-safe design, sharing of SSCs and allowing for equipment outages.” The NRC staff’s evaluation of the PRA, which is used to identify LBEs, is described in section 3.1.1 of this SE and addresses the NRC staff’s review of CCFs in accordance with the PRA standard.

Single layer of defense

The applicant deviated from the NEI 21-07 Section 4.2.1.2 guidance in that the PSAR does not contain an affirmative statement that there is no overreliance on a single layer of defense. However, because there are PSFs that were specifically elevated to NSRST classification for DID adequacy to minimize challenges to the RSFs, and there are PSFs elevated to NSRST classification for DID adequacy and for risk-significance to mitigate consequences of events where the RSFs are assumed to fail, the NRC staff judged that there is not overreliance on a single layer of defense for the LMGS preliminary design.

4.2.1.3 Single Feature Reliance

The NRC staff evaluated the information in PSAR section 4.2 against the following criteria from Section C.4.2.1.3 of NEI 21-07, as endorsed by RG 1.253, to ensure that the LMGS design does not overly rely on any single feature across multiple layers of defense for any risk-significant LBE:

- The applicant should include an affirmative statement that the evaluation of the dependence on a single feature across the layers of defense was performed for all risk-significant LBEs and DBAs.

- The applicant should provide a brief discussion of the method used to evaluate this DID topic.

- The applicant should identify any special treatments added to provide assurance against overreliance on any single feature across multiple layers of defense for any risk-significant LBE.

- The description should be sufficient to identify the DID objectives requiring actions to provide assurance against overreliance on any single feature across multiple layers of defense for any risk-significant LBE. The details of this evaluation should be retained in plant records.

PSAR section 4.2.1.3 and associated PSAR table 4.2-7 summarize the applicant's evaluation of the dependence on a single feature across the layers of defense. The applicant followed a three-step approach that involved evaluating risk-significant DBEs and BDBEs, assessing common dependencies in these events and ensuring there is no undue reliance on a single barrier or feature, and, if undue reliance on a single barrier or feature is identified, taking action to reduce or eliminate it. Because the fuel plays a significant role in plant safety for LMGS, the applicant's evaluation focused on ensuring there was not undue reliance on the fuel. In the evaluation, the applicant first identified LBEs where RSF 1.2.1 for passive heat removal provides a preventive function, because these events have the most potential for overreliance on the fuel particles and pebbles due to the role the fuel plays in heat generation, heat removal, and provides radionuclide retention. These events were used to identify areas needed for more detailed analyses to reduce unnecessary conservatism. The full set of analyzed LBEs was then evaluated to identify if there is a feature that all LBEs rely on to meet the quantitative objectives. The applicant concluded, based on this evaluation, that there was no single feature reliance for the set of risk-significant LBEs.

As discussed in section 5.2 of this SE, radionuclide retention in the fuel is dependent on the success of four RSFs under RSF 1 (control reactivity, control heat removal, control water/steam ingress, maintain geometry), each of which is responsible for controlling different ways the fuel can exceed acceptable limits, thereby failing the radionuclide retention function. Because failure of the fuel's radionuclide retention function may occur if one of the four supporting RSFs fails, it follows that if all supporting RSFs are successful, then the radionuclide retention is not challenged. With respect to single feature reliance, if none of the supporting RSFs rely on a single feature, then by extension there is not single feature reliance on the fuel. As noted in the layer 2 evaluation above, LME selected one or more NSRST function to support each RSF and elevated them to NSRST for ensuring DID adequacy. Additionally, PSF 2 ensures that the HPB appropriately retains radionuclides. The NRC staff finds the applicant's approach to evaluating single feature reliance acceptable for the CP application because this approach ensures that no single feature reliance exists for the current scope, which is limited to in core events. The NRC staff will review considerations for single feature reliance for ex-core events in a future OL application.

4.2.1.4 Prevention-Mitigation Balance

The NRC staff evaluated the information in PSAR section 4.2 against the following criteria from Section C.4.2.1.4 of NEI 21-07, as endorsed by RG 1.253, to ensure that the applicant performed an evaluation of the prevention-mitigation balance for the LMGS design:

- The applicant should affirmatively state that the evaluation of prevention-mitigation balance across the layers of defense was performed for all risk-significant LBEs and DBAs.

- The applicant should provide a brief discussion of the method used to evaluate the prevention-mitigation balance for its design.

- The applicant should provide separate discussions of plant capability and/or programmatic actions added as a result of the prevention-mitigation balance attribute evaluations.

- The description should be sufficient to identify the DID objectives requiring actions to maintain prevention-mitigation balance. The details of this evaluation should be retained in plant records.

PSAR section 4.2.1.4 summarizes the applicant's evaluation of prevention-mitigation balance. The applicant considered prevention and mitigation balance to be sufficient when the IDP confirmed no single design or operational feature is excessively relied upon for the LBEs evaluated for prevention-mitigation balance. The applicant also evaluated prevention-mitigation balance for DBAs by considering functions that prevent the DBEs mapped to each DBA. PSAR table 4.2-7 provides the results of the evaluation, which was performed on an LBE-specific basis, and contains a list of the PSFs that prevent and mitigate each evaluated LBE.

The NRC staff disagrees with LME that evaluating single feature reliance for LBEs is sufficient to demonstrate that there is adequate prevention and mitigation balance, because ensuring more than one function is available does not ensure the functions are balanced between prevention and mitigation. The NRC staff additionally finds that the LBE-specific evaluation provided in PSAR table 4.2-7 appears reasonable, but the evaluation provides a PRA-centric view of prevention that does not address the issue holistically. This method of assessing what SSCs are preventive versus mitigative based on the PRA success or failure, is not consistent with how the prevention and mitigation has traditionally been assessed. The prevention and mitigation balance should look at each function and identify whether its role is to preclude a release from occurring (i.e., prevention function) or to reduce the consequences once a release does occur (i.e., mitigative function), consistent with the NEI 18-04 definitions for prevention and mitigation functions. As discussed in NEI 18-04, SSCs may perform both preventive and mitigative functions. Looking at each function to identify its role as a prevention function or a mitigation function helps address the key safety principle that protection should not rely solely on prevention or solely on mitigation.

The applicant's evaluation does not consider the balance of prevention and mitigation across sequences or LBEs with the same or similar initiating events; this grouping is necessary to clearly address prevention-mitigation balance since lower-frequency LBEs may have many failed functions. Additionally, the applicant's approach does not address any prevention that is

not modeled in the PRA (i.e., design features or SSCs that prevent the occurrence of initiating events outright). These features are essential to the safe design of the facility and should be considered in evaluating prevention-mitigation balance. However, despite the applicant's limited evaluation described above, based on the NRC staff's review of the LBEs discussed in PSAR chapter 3, "Methodologies, Analyses, and Licensing Basis Events," and the functions described in PSAR chapter 5, "PRA Safety Functions, Design Criteria, and SSC Classification," the NRC staff determined that there was adequate prevention-mitigation balance in the design for the CP stage based on the presence of multiple functions to initiate a reactor trip and perform the heat removal function to prevent release from the fuel along with the functional containment layers in the fuel and the additional mitigation provided by the HPB. The NRC staff expects that a more comprehensive discussion of prevention-mitigation balance, consistent with the discussion above, will be provided for review at the OL application stage.

4.2.1.5 Integrated Decision-making Process and Integrated Decision-making Process Panel

The PSAR describes that the applicant used the LMP methodology, as described in NEI 18-04 and endorsed by RG 1.233, to establish a structured risk-informed, and performance-based framework for the selection of LBEs, classification of safety functions, and determination of the appropriate classification and performance requirements for SSCs. One of the key features of the LMP methodology is the use of the IDP, implemented through a multidisciplinary IDPP, to ensure that the risk-informed, performance-based approach is applied consistently and appropriately to establish special treatments and evaluate the adequacy of DID. NEI 18-04 states that the IDP should be conducted by a knowledgeable, multidisciplinary panel, and that the process should be traceable, documented, and consistent with the LMP framework. The DID evaluation ensures that plant design and operational strategies provide multiple, independent, and layered means to protect public health and safety, consistent with NRC safety philosophy and the Commission's safety goal policy statement.

The NRC staff reviewed PSAR section 4.2, which includes the applicant's descriptions of the IDP and IDPP implementation. The NRC staff audited supporting documentation to confirm appropriate implementation of the IDP and IDPP. The NRC staff determined that the applicant's IDP approach is consistent with NEI 18-04 as endorsed by RG 1.233, and that the IDPP deliberations resulted in defensible, traceable decisions that support the preliminary licensing basis for the CP stage as discussed below.

The NRC staff reviewed the applicant's description of the IDP evaluation of LBEs, including DBAs, and the associated limitations of the analyses, assumptions, and uncertainties. The applicant acknowledged the IDP's role in integrating deterministic and probabilistic approaches, which the applicant states ensure that all relevant scenarios are evaluated, limitations are acknowledged and managed, and safety margins are maintained using conservative assumptions. The NRC staff conducted a detailed assessment of the applicant's approach to LBE evaluation in chapter 3 of this SE, which was found to be acceptable for the CP stage.

The NRC staff reviewed the applicant's description of the composition of the IDPP as a diverse group of experts covering important process and functional elements of the organization, following a structured process for documenting IDPP meetings and decisions. The PSAR states that the IDPP's primary responsibility is to review and approve SSC classifications, and the IDPP's decisions support the development, modification, and confirmation of design requirements, including quality assurance requirements, to appropriately reflect SSC safety

significance and classification. The NRC staff conducted a detailed assessment of SSC classifications in chapter 5 of this SE, where they were found to be acceptable for the CP stage.

As captured in RCI-1 and discussed in section 5.1 of this SE, LME added a step to their SSC classification process in which the results of an evaluation of SSCs identified as significant risk contributors through an evaluation that assumes failure of the SSC and compares the resulting cumulative risk metrics to their respective limits are provided to the IDPP for their determination on whether the SSCs should be classified as risk-significant as part of the SSC safety classification process. This step was not fully incorporated into the CP application. The implementation of this step will be reviewed as part of a future OL application review.

The NRC staff also reviewed the applicant's description stating that the IDP is used to establish the plant DID capabilities in accordance with NEI 18-04, Section 5.7.2, which provides questions to focus the IDP on the evaluation of LBEs. The IDPP also reviews and approves both plant capability and programmatic DID, including reliability and capability targets and special treatments. The NRC staff reviewed the applicant's discussion of plant capability DID adequacy in PSAR section 4.2 and determined that it reflected consideration by the IDPP. While programmatic DID is not fully addressed in the PSAR as discussed in section 4.2.1 of this SE, the NRC staff assessed the applicant's approach to reliability and capability targets in chapters 6 and 7 of this SE for SR and NSRST SSCs, respectively, where they were found to be acceptable for the CP stage.

Finally, the applicant noted that, for the FSAR, the IDP and IDPP must be proceduralized and controlled, formal training and qualification will be required for IDPP members, special treatments will be finalized and detailed to the component level for SR and NSRST SSCs, and programmatic special treatments will be developed to ensure reliability and capability targets will be met. Because the IDP and IDPP were sufficiently implemented to develop preliminary evaluations for the PSAR, the NRC staff finds them to be acceptable for a CP application. The NRC staff will review the applicant's final implementation of IDP and IDPP in a future OL application.

Based on its review of the PSAR, the NRC staff finds that LME's implementation of the IDP and IDPP for the LMGS application is consistent with the methodology described in NEI 18-04 as endorsed by RG 1.233. The NRC staff finds that the IDPP composition is appropriately diverse and the IDP is implemented to ensure balanced consideration of probabilistic and deterministic approaches while allowing engineering judgment to be appropriately exercised. Therefore, the NRC staff concluded that LME's IDP and IDPP implementation provides an acceptable basis for supporting the CP application in accordance with 10 CFR 50.34(a)(1) and 10 CFR 50.34(a)(4) and 10 CFR 50.35, consistent with the NRC-endorsed LMP methodology.

4.2.2 Conclusion

Based on the information provided in PSAR section 4.2, the NRC staff finds that the LMGS design includes DID sufficient to meet 10 CFR 50.34(a)(1) and 10 CFR 50.34(a)(4). This finding is based on the applicant's completion of the appropriate plant capability DID evaluations to an extent that is acceptable for a CP, as documented in sections 4.2.1.1 through 4.2.1.4 of this SE, as well as the applicant's implementation of the IDP and IDPP for the CP application, as documented in section 4.2.1.5 of this SE.

5 PROBABILISTIC RISK ASSESSMENT SAFETY FUNCTIONS, DESIGN CRITERIA, AND STRUCTURE, SYSTEM, AND COMPONENET CLASSIFICATION

The Long Mott Generating Station (LMGS) preliminary safety analysis report (PSAR) chapter 5, "PRA Safety Functions, Design Criteria, and SSC Classifications," describes the LMGS probabilistic risk assessment (PRA) safety functions (PSFs), including required safety functions (RSFs), and required functional design criteria (RFDC), complementary design criteria (CDC), principal design criteria (PDC), and classification of safety-related (SR) and non-safety-related with special treatment (NSRST) structures, systems, and components (SSCs).

Long Mott Energy, LLC. (LME) states in PSAR chapter 5 that: "[t]he Xe-100 design follows the methodology in Nuclear Energy Institute (NEI) 18-04 as described in the Xe-100 Risk-Informed Performance-Based Technology Inclusive Licensing Basis Development Licensing Topical Report (TR) (ML22187A279)." This safety evaluation (SE) chapter evaluates the consistency of LME's methodology and results of SSC classification with the approach outlined in NEI 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Revision 1 (ML19241A472) and the TR.

PSAR table 1.4-4 incorporates by reference (IBR) the TR, which the U.S. Nuclear Regulatory Commission (NRC) staff approved in an SE dated August 4, 2022 (ML22187A279). PSAR table 1.4-6 addresses the one limitation documented in the NRC staff's SE of this TR (ML22187A271) by noting that exemptions from the regulations related to LME's implementation of the NEI 18-04 methodology are addressed in Part V of the construction permit (CP) application.

The NRC staff's evaluation of the exemption from the "safety-related SSCs" definition under Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.2, "Definitions," and 10 CFR 26.5, "Definitions," and "safety-related electric equipment" defined in 10 CFR 50.49(b)(1), is documented in section 15.2 of this SE.

5.1 Safety Classification of Structures, Systems, and Components

PSAR section 5.1 describes the SSC classification methodology and categories for the LMGS design. The SSC classification categories are SR, NSRST, and non-safety-related with no special treatment (NST).

PSAR section 5.2 provides a high-level description of the PSFs that were identified as SR (also referred to as RSFs) and NSRST through the NEI 18-04 process. PSAR section 5.4 provides the selection of the SR SSCs performing RSFs and PSAR section 5.5 provides the selection of NSRST SSCs to perform NSRST PSFs. PSAR chapter 6, "Safety-related SSC Criteria and Capabilities," and PSAR chapter 7, "Non-Safety-Related with Special Treatment Structures, Systems, and Components Criteria and Capabilities," provide additional details on the criteria and capabilities of safety-significant SSCs (i.e., those classified as SR and NSRST).

The applicable regulatory requirements for the evaluation of SSC classification are as follows:

- 10 CFR 50.34(a)(1); and
- 10 CFR 50.34(a)(4).

The applicable guidance for SSC classification is as follows:

- Regulatory Guide (RG) 1.233, Revision 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 NEI 18-04, Revision 1, with clarifications.

- RG 1.253, Revision 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 NEI 21-07, Revision 1, "Technology Inclusive Guidance for Non-Light Water Reactors, Safety Analysis Report Content: For Applicants Using the NEI 18-04 Methodology," (ML22060A190), with clarifications and additions.

The regulations in 10 CFR 50.34(a)(1) and (4) together require that each CP applicant submit, in its PSAR, a safety assessment with a preliminary analysis and evaluation of the design and performance of facility SSCs. Specifically, 10 CFR 50.34(a)(1)(ii)(D) requires consideration be given to the safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. The requirements of 10 CFR 50.34(a)(4) specify that the objective of the preliminary analysis and evaluation of the design and performance of SSCs is to assess the risk to public health and safety resulting from operation of the facility, including determination of the margins of safety during normal operations and transient conditions during the life of the facility, and the adequacy of SSCs provided for the prevention of accidents and mitigation of consequences of accidents. These required assessments of the facility provide an understanding of those SSCs that are most important to radiological health and safety because they play a key role in preventing and mitigating accidents. The regulations in 10 CFR Part 50, "Domestic Licensing Production and Utilization Facilities," establish that certain SSCs are SR (as defined in 10 CFR 50.2) and require applicants and licensees to design, fabricate, construct, and test those SSCs in accordance with standards, quality requirements, etc., beyond those applied in normal industrial practice to commercial grade SSCs as necessary to ensure their reliability and capability to prevent and mitigate accidents; other requirements may apply to SSCs that are safety-significant but not SR.

The guidance in NEI 18-04, as endorsed with clarifications in RG 1.233, provides an acceptable process for using the PRA and licensing basis event (LBE) analyses to classify an SSC based on the role it plays in prevention and mitigation of LBEs, as described in more detail in PSAR chapter 3, "Methodologies, Analyses, and Licensing Basis Events" and chapter 3 of this SE. Under this process, the most safety-significant SSCs are classified as SR, SSCs that perform safety-significant functions but are less safety-significant are classified as NSRST, and other SSCs are classified as NST. These classifications are an integral part of the licensing basis development process described in NEI 18-04, because they are used to assess SSCs according to their safety or risk significance and identify if special treatments beyond normal industrial practices are needed to ensure SSC performance of safety functions in the prevention and mitigation of LBEs.

5.1.1 Technical Evaluation

The NRC staff reviewed PSAR section 5.1 to determine if the SSC classification methodology and categories conform to the relevant guidance in RG 1.233 and RG 1.253. The NRC staff also held audit discussions focused on LME's implementation of the methodology described in the PSAR to confirm that it is consistent with the NEI 18-04 methodology and the RGs, as documented in the audit report (ML26293A340).

During its review of the PSAR, the NRC staff identified certain areas of the NEI 18-04 guidance and corresponding endorsement in RG 1.233 that were interpreted differently by LME and the NRC staff. Through audit discussions, understanding was reached on these differing interpretations and how they may have impacted the SSC safety classifications provided in the PSAR. In areas where LME had unintentionally deviated from the licensing modernization project (LMP) methodology, LME implemented changes to its process to align more closely with the LMP methodology. One such area was captured in the NRC staff's request for confirmation of information (RCI)-1 dated August 14, 2026 (ML26230A155) and in the evaluation below.

RCI-1 documents preliminary results of the risk-significance evaluation steps that were inadvertently not performed by LME. As confirmed through RCI-1, these results are not incorporated into PSAR Revision 1, but at the time of this SE are in the process of being incorporated into the safety analysis and design. The NRC staff has documented areas where the design and special treatments are expected to change based on these preliminary results and confirmation of final risk-significance determinations and appropriate treatment within the design will be performed at the operating license (OL). The PSAR currently includes results that are inconsistent with the results application of the methodology as revised in RCI-1 (see table 5.1-2 below). Application of the methodology described in RCI-1 will yield different results, as described below, based on corrections to the methodology. To the extent use of the corrected methodology provides confidence that the final risk-significance determinations will be satisfactory, that information will be available for the OL application and the NRC staff will review it in the context of the OL application.

The NRC staff reviewed PSAR section 1.3.1, which states that the applicant follows the overall NEI 18-04 methodology, as endorsed in RG 1.233, and PSAR section 5.1.1, which states that LME follows the risk-informed and performance-based methodology in RG 1.233. Classification of SSCs is a key component of the NEI 18-04 methodology and based, in part, on the safety functions the SSC performs and the SSC's safety significance. LME describes a systematic and iterative process for selection of LBEs, classification of SSCs, and determination of defense-in-depth (DID) adequacy. As described in PSAR section 5.1.1, LME follows the guidance in RG 1.233 with some departures in the classification of essential support SSCs as discussed below.

Consistent with the guidance in RG 1.233, LME uses the following SSC classification categories:

- Safety-related (SR):
    - SSCs selected by the designer from the SSCs that are available to perform the RSFs to mitigate the consequences of design-basis events (DBEs) to within the LBE Frequency-Consequence (F-C) Target, and to mitigate design-basis accidents

(DBAs) that only rely on the SR SSCs to meet the dose limits of 10 CFR 50.34, "Content of applications; technical information," using conservative assumptions.

- SSCs selected by the designer and relied on to perform RSFs to prevent the frequency of beyond-design-basis events (BDBEs) with consequences greater than the 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the F-C target.

- Non-safety-related with special treatment (NSRST):
  - Non-safety-related SSCs relied on to perform risk-significant functions. Risk-significant SSCs are those that perform functions that prevent or mitigate any LBE from exceeding the F-C target or make significant contributions to the cumulative risk metrics selected for evaluating the total risk from all analyzed LBEs.
  - Non-safety-related SSCs relied on to perform functions requiring special treatment for DID adequacy.

- Non-safety-related with no special treatment (NST):
  - All other SSCs (with no special treatment required).

Table 5.1-1 of this SE summarizes the six evaluation steps in the NEI 18-04 SSC methodology that are used in determining what SSCs fall into each of the SSC categories. The first five evaluation tasks are quantitative steps performed using the PRA. The final step, the DID evaluation, is a qualitative step that assesses the layers of safety in the design and ensures the SSC classification is risk-informed. The DID evaluation is described in PSAR section 4.2 and evaluated in section 4.2 of this SE.

During its review, the NRC staff determined that LME's SSC classification methodology was inconsistent with the NEI 18-04 methodology as listed below. As described in the evaluation that follows, the NRC staff evaluated the safety impact of these inconsistencies at the CP application stage. For tasks 7a.1 and 7a.2, the PSAR results are consistent or conservative with what would have resulted from following the NEI 18-04 methodology. For tasks 7a.3 and 7b, LME confirmed via RCI-01 changes to its evaluation that will be implemented but were not incorporated into the PSAR.

- LME assumed failure of all non-safety-related SSCs instead of assessing the impact of individual functions or SSCs and did not include BDBEs with frequency bands extending into the DBE region in the evaluation for task 7a.1 in table 5.1-1 of this SE.

- LME did not define the process for evaluating BDBEs that exceeded the 10 CFR 50.34 dose limits (i.e., high-consequence BDBEs) to assess which SSCs were relied on to maintain LBEs below the F-C target as described in task 7a.2 in table 5.1-1 of this SE.

- LME did not perform task 7a.3 in table 5.1-1 of this SE, but the evaluation performed in lieu of task 7a.1 might have assigned a SR classification to some of the impacted SSCs and others were assigned NSRST for DID under step 7e.

- LME's evaluation did not assume failure of the function or SSC and the impact on the cumulative risk metrics as described in task 7b in table 5.1-1 of this SE.

- LME identified an alternative approach for the classification of some load path SSCs for SSCs performing SR functions from that in RG 1.233, NRC staff position C.2.a. Load path SSCs for SSCs performing some SR functions are classified as risk-significant NSRST.

Task 7a.1

Instead of performing evaluations for individual function or SSC impacts on LBEs, LME performed an evaluation that assumed failure of all non-safety-related (i.e., NSRST and NST) SSCs and showed that no anticipated operational occurrences (AOOs) or DBEs exceeded the F-C target at 95th percentile frequency and consequence. The NRC staff determined that LME's evaluation produces results generally expected to be conservative compared to the task 7a.1 evaluation because of the use of the 95th percentile instead of the mean and the evaluation of all AOOs instead of only those AOOs that extend into the DBE region. However, because BDBEs with 95th percentiles extending into the DBE region were not included in the evaluation, it is possible, though unlikely, that this preliminary classification process did not identify an RSF that would have been identified with the inclusion of these BDBEs.

The NRC staff also determined that because all functions and SSCs were evaluated together, the risk insights that are gained through the individual function or SSCs evaluations were not available. The NRC staff determined it to be acceptable that these risk insights are not available at the CP stage because the generally conservative approach taken to identify SR SSCs and the approach to identify DID NSRST SSCs (discussed below and in section 4.2 of this SE) are unlikely to miss risk insights that would necessitate design changes. Additionally, LME has made corrections to its methodology to perform these evaluations as confirmed in the response to RCI-1 and will incorporate these risk insights in its design.

Task 7a.2

For task 7a.2, LME did not identify any BDBEs within the preliminary set of LBEs that exceeded 10 CFR 50.34 dose limits, referred to as high-consequence BDBEs by NEI 18-04. Because there were no high-consequence BDBEs, not having the evaluation against the F-C target defined within the process did not impact the results of the preliminary SSC classification. As the PRA is expanded beyond full power operation core events, the LBE set may include high-consequence BDBEs; LME has updated its methodology to ensure the impact of failed functions or SSCs on these LBEs is evaluated consistent with the NEI 18-04 methodology and the description of SR SSCs in PSAR section 5.1. The NRC staff has accordingly determined that the updated methodology is adequate to identify SR SSCs from these high-consequence BDBEs.

Task 7a.3

The NRC staff's review of the evaluation of AOOs and DBEs described under task 7a.1 above would have assigned a classification of SR instead of NSRST to SSCs where the modeled failure of the SSC would cause the AOO or DBE results to exceed the F-C target at the 95th percentile, but not at the mean. The NEI 18-04 methodology indicates that these SSCs can be

classified as risk-significant NSRST. The risk-significance information to determine if any SSCs were conservatively assigned SR instead of risk-significant NSRST through this process was not available at the CP stage because, as noted above, LME did not perform an evaluation to determine the impacts of individual function or SSC failing on the LBEs in support of the PSAR. However, during the review, LME updated its SSC classification methodology to include these evaluations and performed a preliminary evaluation that included an assessment of BDBEs that were not covered by the 7a.1 evaluation. LME described the preliminary results of this evaluation in the response to RCI-1 dated August 28, 2026 (ML26239A157). The applicant's preliminary evaluation identified 14 risk-significant systems that keep LBEs below the F-C target. Table 5.1-2 of this SE provides this list of risk-significant systems. Of the 14 SSCs identified in the preliminary list of risk-significant SSCs, 13 are identified as SR or NSRST within the PSAR. The remaining system, the conventional island electrical system, that is currently classified as NST in PSAR Revision 1 but included in the preliminary list has a reasonable likelihood of being classified NST with further refinement of the PRA and design, as discussed in more detail in sections 5.2 and 5.5 of this SE. Table 5.1-2 also includes the impact on the other system classifications and the sections of the SE where the aspects of the systems and their functions are further evaluated.

Task 7b

An area of NEI 18-04 that was interpreted differently by LME and the NRC staff was the evaluation against the cumulative risk targets, task 7b. Key statements from NEI 18-04 relating to this evaluation include:

- "In this task, the integrated risk of all the LBEs, is evaluated against three cumulative risk targets… One element of this task is to identify design features that are responsible for preventing and mitigating radiological releases and for meeting the integrated risk criteria."

- "SSCs are classified as risk-significant if the SSC function is necessary to keep any LBEs inside the F-C Target, or if the total frequency of LBEs with the SSCs failed is within 1% of any of the three cumulative risk targets identified in Task 7b."

- "This approach to assigning risk significance uses the concept of evaluating the impact of the SSC function on the ability to meet the F-C Target, and also includes criteria based on risk significance metrics for the cumulative risk impacts of SSC functions across all the LBEs."

- "The safety classification process and the corresponding special treatments serve to control the frequencies and consequences of the LBEs within the F-C Target and to ensure that the cumulative risk targets are not exceeded."

The NRC staff's understanding is that this evaluation is to be performed similar to the LBE evaluation, by first assuming failure of the function or SSC and then evaluating the impact on the cumulative risk metrics. The NRC staff's understanding is based on the passages above, which indicate that the purpose of the evaluation is to measure the function or SSC's role in maintaining cumulative risk within the target and that the cumulative risk evaluation is similar to the LBE evaluation but looking across all LBEs.

LME's interpretation of this evaluation is that no additional failures should be taken, and the frequency metric used should be the total frequency for all LBEs where the SSC was already failed in the baseline model. This approach, however, measures something different than the LBE evaluation; it does not measure how important the function or SSC is in keeping the cumulative risk within the F-C target, but rather measures the baseline contribution to the cumulative risk. The applicant's approach identifies where additional margin to cumulative risk could be gained if the failure frequency were reduced rather than cumulative contribution to keeping LBEs below the F-C target.

Following the discussion in the regulatory audit, LME adjusted its SSC classification methodology to add a step in assessing a function or an SSC's impact on the cumulative risk metrics. This step is confirmed in the response to RCI-1 and includes setting the failure probability for each function or SSC to 1.0 and then comparing the results to 100 percent of each of the cumulative risk metrics. When one or more of the cumulative risk metrics are exceeded, the function or SSC will be identified as a significant risk contributor. This will not automatically change the classification, but the information will be provided to the integrated decision-making process panel (IDPP) to determine whether the classification should be adjusted. This assessment is performed to meet the non-light-water reactor PRA Standard Table 1.9-2, but LME included the review of this information by the IDPP as part of its formal process to address the difference in interpretation for a function or SSC's impact on the cumulative risk metrics.

The NRC staff determined this approach to be acceptable because it provides a reasonable way of assessing the role an SSC plays in maintaining cumulative risk within the target. As discussed above, the NRC staff's conclusion has two bases. First, LME adjusted the methodology to be consistent with the guidance in NEI-18-04, as endorsed in RG 1.233, which provides confidence that the final risk-significance determinations will be satisfactory. Second, preliminary application of the adjusted methodology does not change the results for 13 out of 14 systems and indicates that the status of the 14th system, the conventional island electrical system, currently identified as NST, will not likely change. The NRC staff will review the final results of both cumulative risk evaluations in the OL application.

Task 7d

Another difference between LME's preliminary methodology and NEI 18-04 that was identified during the review was in the assessment of LBEs in different LBE categories based on their uncertainty bands. While LBEs are generally categorized based on their mean frequency, the methodology in NEI 18-04 indicates that if the uncertainty band straddles a frequency boundary that the LBE be evaluated in both categories. This impacts the LBE evaluations for the identification of RSFs, as discussed above, and the identification and evaluation of DBAs. In addition to identifying DBAs for DBEs, DBAs should also be identified for AOOs and BDBEs with frequency bands that extend into the DBE region, potentially leading to additional RSFs being identified. Because PSAR Revision 0 did not include these additional events, when LME performed the mapping of these additional events to DBAs some events did not have a corresponding DBA and were mapped to the seismic event. This is acceptable for the CP stage as the dose would bound the appropriate DBAs for these events and the RSFs are assessed through the full scope of DBA analyses.

LME added the mapping of these events to DBAs in PSAR Revision 1. In the process, LME identified two potential impacts to the design. First, LME identified SSGTL-29, a steam

generator tube leak (SGTL) with concurrent failure of the steam generator isolation valves, as not being bounded by the current DBAs and having a potentially unacceptable volume of water entering the reactor. LME addressed this by removing some of the conservatism from the frequency of the initiating event for this LBE within the PRA. The NRC staff finds this to be reasonable based on the data for steam generator tube failure discussed in PSAR section 6.4.12 and the wide uncertainty bands used. LME identified a second impact related to pressure relief on the steam generator, where the open function was upgraded to SR because of one of the steam generator tube rupture events being mapped to a DBA. Both LME's implementation of the NEI 18-04 methodology and the results in the PSAR were updated to address these differences identified in the regulatory audit.

During the reevaluation of DBA mapping, LME identified some LBEs where the DBA with a similar event sequence was not bounding due to the failure of a SR SSCs as part of the DBE or BDBE. For the CP application, LME has mapped these events to the seismic DBA, which the NRC staff finds acceptable based on the bounding nature of this event. Consistent with NEI 18-04 and LME's SSC classification methodology, LME will provide DBAs consistent with its methodology for mapping LBEs in an OL application.

RG 1.233, NRC Staff Position C.2.a

LME describes its approach for classification of some load path SSCs in PSAR section 5.1.1. As stated in RG 1.233, NRC staff position C.2.a: "The staff expects that SSCs that provide essential support (including required human actions) for SR or NSRST SSCs will be classified in a manner consistent with the higher-level function, even if the supporting SSC is not explicitly modeled in the PRA." This includes SSCs that provide a load path for SSCs performing a safety-significant function.

PSAR section 5.1.1 describes an alternative approach for classification of load path SSCs providing essential support. This alternative approach is applied to SSCs that do not perform safety-significant functions other than essential support and does not include SSCs that contribute to RSF 1.4, "Maintain Geometry," which are classified as SR based on their contributions to performing this SR function. (These load path SSCs provide structural support but not essential support.) However, LME classifies other load path SSCs providing structural support for SSCs performing SR functions as risk-significant NSRST.

As described in PSAR section 3.7, these SSCs are assigned to seismic category IIA with a **[[ ]]** For SSCs that provide a load path for NSRST SSCs, the classification and **[[ ]]** will conform to that of the supported SSC unless the seismic PRA (SPRA) demonstrates that acceptable performance can be achieved with reduced requirements.

The NRC staff determined that this approach is acceptable based on the following:

- Load path SSCs that contribute to maintaining core geometry are SR. These SSCs provide structural support for the reactor system, fuel in the reactor system, SR portions of the reactor cavity cooling system, and the control rods and control rod drives that are the primary SSCs for addressing functional containment, control of heat generation, and control of heat removal.

- As noted in section 3.7 of this SE, the NRC staff determined that the associated design requirements applied should achieve performance comparable to SR SSCs at the safe shutdown earthquake and undergo only **[[ ]]** Accordingly, a load path SSC will not impact the ability of a connected SR SSC to perform its function.

- LME plans to perform a confirmatory SPRA at the OL application stage, as stated in PSAR section 3.7, which would provide further confidence that this alternative approach to classification will achieve a comparable level of performance and safety for the facility.

- A graded approach to quality assurance (QA) is applied to risk-significant NSRST SSCs, as described in PSAR section 8.1 and in the LMGS QA Program Description (ML25269A117). Through this program, QA measures are applied as special treatments, targeted at those characteristics or critical attributes that make them NSRST.

- Construction fitness-for-duty applies to these SSCs based on the definition of constructing and construction activities in 10 CFR 26.5, which states: "tasks [involved in building a nuclear power plant performed at the location where it will be operated] include fabricating, erecting, integrating, and testing safety- and security-related SSCs, and the installation of their foundations, including the placement of concrete."

Overall SSC Classification Methodology

The SSC classification in the PSAR is based on the preliminary LMGS design and associated LMP process implementation including the PRA discussed in PSAR section 3.1.1. LME committed to follow the methodology described above, which is based on the LMP process in NEI 18-04 and RG 1.233, and the NRC staff will review the finalized SSC classification described in the OL application when the LMGS design is mature and the LMP process is complete. As discussed in section 3.1.4 of this SE, there were limited consequence analysis results generated by LME for use in the PSAR. This is expected at the CP stage but impacts the completeness of the SSC classification. For example, when looking at initiating event families, it is expected that each subsequent failure within an initiating family set following a release would result in increased consequences, because the conclusion otherwise is that the function or SSC has no mitigative impact. However, with approximately 30 unique consequence results and over 280 events, there are many subsequent failures within the LME LBE set where the impacts are unknown because unique consequence results were unavailable. As explained in section 3.1.4 of this SE, the same bounding consequences were assigned to multiple LBEs within the set, which is adequate for a CP application.

As stated in PSAR section 3.2 and discussed in section 3.2 of this SE, LME did not perform the LBE family grouping step described in NEI 18-04. The family grouping step identifies the LBEs where an SSC has little or no impact on the likelihood of an event occurring or the mitigation of that event and provides for grouping those LBEs into a family such that they are represented by the LBE family and not modeled as a unique event within the LBE set. While modeling of individual LBEs without family groupings makes safety classification steps more challenging, it is not required to determine appropriate safety classification of SSCs. Therefore, the omission of the family grouping step is acceptable at the CP stage. LME states in PSAR section 3.2.4 that it will perform the family grouping step in determining the final LBE set for the OL application. As stated in PSAR section 3.1.4 and discussed in section 3.1.4 of this SE, for the final set of LBEs,

LME will provide a unique source term in the final safety analysis report (FSAR) to demonstrate each function's or SSC's impact on event frequency and consequence.

Based on its review as described above, the NRC staff finds that LME's process for establishing the set of LBEs and classifying SSCs follows the risk-informed and performance-based methodology in RG 1.233 with the few exceptions evaluated above, all of which the NRC staff determined to be acceptable. Accordingly, the NRC staff finds that the SSC safety classification categories are consistent with RG 1.233. The NRC staff has determined that the level of detail provided in the applicant's PSAR is consistent with the guidance in NEI 21-07, as endorsed with clarifications in RG 1.253, given the preliminary design at the CP stage.

Consistent with the methodology in RG 1.233, the process for finalizing the classification of SSCs will continue as the design matures and the implementation of the systematic and iterative methodology in RG 1.233 progresses.

5.1.2 Conclusion

Based on the considerations described above, the NRC staff finds that the SSC classification categories and process described in PSAR section 5.1 is consistent with the applicable regulatory guidance with the limited exceptions evaluated above and determined to be acceptable and, therefore, meets the regulatory requirements in 10 CFR 50.34 identified above for the issuance of a CP in accordance with 10 CFR 50.35, "Issuance of Construction Permits."

Table 5.1-1: SSC Classification Steps in NEI 18-04 Methodology

Task[1]	LBEs Evaluated	Evaluation	Criteria Compared Against	Values Used	Resulting SSC Classification and Risk Category
7a.1	DBEs and AOOs and BDBEs with uncertainty bands that extend into the DBE region	Remove individual functions and evaluate impact on individual LBEs	F-C target	Mean	SR RSF, risk-significant and safety-significant
7a.2	BDBEs with mean consequences that exceed 25 rem (0.25 Sv) TEDE	Remove individual functions and evaluate impact on individual LBEs	F-C target	Mean	SR RSF, risk-significant and safety-significant
7d	DBAs	Conservative consequences evaluation in which only available SR SSCs perform their functions	10 CFR 50.34 dose limits	95th percentile or greater	SR RSF, safety-significant
7a.3	All non-DBA LBEs	Remove individual functions and evaluate impact on individual LBEs	F-C target	95th percentile	NSRST, risk-significant and safety-significant
7b	All non-DBA LBEs	Remove individual functions and evaluate total frequency of all associated LBEs[2]	1% of cumulative risk metrics	Mean	NSRST, risk-significant and safety-significant
7e	All non-DBA LBEs	Perform DID adequacy evaluation using the integrated decision-making process (IDP) and IDP panel (IDPP)	Table 5-2 and other DID concepts described in section 5 of NEI 18-04	N/A	NSRST, safety-significant

1. From section 3.2.2 of NEI 18-04. Note, task 7c of determining risk significance is an assessment of the results determined in steps 7a and 7b.
2. LME's interpretation of this step in the NEI 18-04 methodology is that the total frequency of all associated LBEs where the SSC is modeled as failed in the baseline PRA model is compared to 1% of the cumulative risk metrics. See additional discussion on this aspect in the evaluation section above.

Table 5.1-2: Impact to Highest Risk Category Classification for System Impacted by RCI-1

System	Preliminary LME Methodology, Reflected in PSAR	Revised Methodology, Preliminary Results in RCI-1	SE Evaluation Section
Conventional island electrical system	NST	NSRST, risk-significant	5.5 and 1.3
Distributed control system	NSRST, DID	NSRST, risk-significant	5.5 and 7.3.12
Helium circulator system	NSRST, DID	NSRST, risk-significant	5.5 and 7.3.5
Investment protection system	NSRST, risk-significant	NSRST, risk-significant	5.5 and 7.3.11
Nuclear island electrical system	NSRST, DID	NSRST, risk-significant	5.5 and 7.3.8
Nuclear instrumentation system	SR and NSRST, risk-significant	SR and NSRST, risk-significant	5.4, 5.5, and 6.4.14
Primary pressure relief system	SR and NSRST, DID	SR and NSRST, DID	5.4, 5.5, and 6.4.9
Passive reactor cavity cooling system	SR	SR	5.4 and 6.4.11
Reactor building cooling water	NSRST, DID	NSRST, risk-significant	5.5 and 7.3.9
Reactor building pressure relief	NSRST, DID	NSRST, risk-significant	5.5 and 6.4.1
Reactivity control and shutdown system	SR and NSRST, DID	SR and NSRST, risk-significant	5.4, 5.5, and 6.4.10
Reactor protection system	SR	SR	5.4 and 6.4.13
Steam generator system	SR and NSRST, risk-significant	SR and NSRST, risk-significant	5.4, 5.5, and 6.4.12
Start-up and shutdown system	NSRST, DID	NSRST, risk-significant	5.5 and 7.3.4

5.2 PRA Safety Functions

PSAR section 5.2 describes the safety-significant PSFs, which include both the SR PSFs (called the RSFs) and the NSRST PSFs. The RSFs are those PSFs credited to maintain the consequence of one or more DBEs or the frequency of one or more high-consequence BDBEs inside the F-C target, and to conservatively ensure that the 10 CFR 50.34 dose requirements can be met. The NSRST PSFs are PSFs that are either risk significant or required for DID adequacy.

The applicable regulatory requirements for the evaluation of safety-significant PSFs are as follows:

- 10 CFR 50.34(a)(1);
- 10 CFR 50.34(a)(4); and
- 10 CFR 50.35.

The applicable guidance for the selection of safety-significant PSFs is as follows:

- RG 1.233, Revision 0, which endorses NEI 18-04 Revision 1, with clarifications.
- RG 1.253, Revision 0, which endorses NEI 21-07, Revision 1, with clarifications and additions.

5.2.1 Technical Evaluation

The NRC staff reviewed PSAR section 5.2 to determine if the identification of safety-significant PSFs is consistent with the methodology described in PSAR section 5.1. The NRC staff's evaluation of the methodology is documented in section 5.1 of this SE. As defined in NEI 18-04, PSFs are those functions responsible for the prevention and mitigation of an unplanned radiological release from any source within the plant. As part of this evaluation, the NRC staff audited internal applicant documents to examine additional details for consistency with information provided in the PSAR.

The PSAR includes a preliminary determination of the safety-significant PSFs. The applicant used a risk-informed and performance-based methodology for assessing safety-significant PSFs based on the results from the PRA. The applicant used the preliminary PRA described in PSAR section 3.1.1 and evaluated in section 3.1.1 of this SE to determine the safety-significant PSFs for the LMGS.

As discussed in PSAR section 5.2, the safety functions modeled in the LMGS CP PRA are limited in scope. The scope of the safety functions described in the PSAR that LME evaluated through the PRA is limited to controlling radiation from the core, because as discussed in PRA section 3.1.1, the PRA scope is limited to the radiological sources within the helium pressure boundary (HPB). Additionally, the PRA scope did not include external hazard events, which LME assessed in the PSAR through the use of a design-basis hazard level and seismic DBA. As discussed in section 3.1.1 of this SE, the NRC staff finds that the limited scope of the PSFs identified in the PSAR is reasonable and consistent with RG 1.233. LME addressed safety

functions outside the scope of the preliminary PRA at a preliminary level using deterministic methods for the PSAR, but these safety functions will be assessed further in future PRA revisions following the iterative RG 1.233 process as discussed in PSAR 1.3.1.

The applicant defined PSFs at either a functional level or at the system level using the preliminary PRA that supports the PSAR. This is also reflected in the high level at which PSFs are discussed within PSAR section 5.2 and limited information available on the component level safety functions at this preliminary design stage. As the PRA is further developed to support the FSAR, the applicant will further refine the PSFs to provide additional insights into the specific subsystems or components that perform the safety-significant and risk-significant functions. The NRC staff will review the final list of PSFs at the OL stage.

RSFs

The NRC staff reviewed the description of RSFs provided in the PSAR. Consistent with NEI 21-07, the PSAR summarizes how the LMGS Xe-100 design RSFs adequately support the fundamental safety functions (FSFs). The FSFs are defined in NEI 18-04 as "Safety functions common to all reactor technologies and designs; includes control heat generation, control heat removal and confinement of radioactive material." The NRC staff also reviewed PSAR section 1.3.2, which provides additional details on FSFs for the LMGS design.

For the LMGS design, RSF 1, "Retain Radionuclides in Fuel Particles and Pebbles," which is the top-level RSF, supports the confinement of radioactive materials FSF. RSF 1.1, "Control Reactivity," supports the control heat generation FSF. RSF 1.2, "Control Heat Removal," supports the control heat removal FSF. Based on the identification of these RSFs for the LMGS design, the NRC staff finds that the FSFs are adequately captured as part of the LMGS RSFs. In addition to the RSFs that align to the FSFs identified in NEI 21-07, the applicant identifies reactor-specific RSFs as follows:

- RSF 1.3, "Control Water/Steam Ingress"; and
- RSF 1.4, "Maintain Geometry."

RSFs 1.1 - 1.4, which support RSF 1, are further divided into detailed subfunction RSFs as follows:

- RSF 1.1.1, "Control reactivity with inherent reactivity feedback," and RSF 1.1.2, "Maintain long-term subcriticality," for RSF 1.1;
- RSF 1.2.1, "Maintain core heat removal through passive means," for RSF 1.2;
- RSF 1.3.1, "Isolate water/steam sources," for RSF 1.3; and
- RSF 1.4.1, "Maintain core geometry," and RSF 1.4.2, "Maintain reactor support geometry," for RSF 1.4.

The NRC staff's review of LMGS approach to defining RSFs identified that the structure of the FSF associated with confinement of radioactive materials being the highest level function with all other functions supporting it is reasonable for a tri-structural isotropic (TRISO) fueled system

because the functional containment is inherent to the fuel. For a TRISO fueled system, the potential failure modes of that functional containment are tied to the FSFs of controlling heat generation and controlling heat removal, and the LMGS RSFs reflect those facts. Further, the reactor-specific RSFs address the remaining two identified failure modes, with oxidation related failure being addressed by RSF 1.3 and mechanical failure being addressed by RSF 1.4.

As the PRA is expanded to include the control of radiation from processes outside the core, the applicant may identify additional radionuclide retention PSFs. The NRC staff has nonetheless determined that the source terms that exist outside of the fuel would be limited compared to the source term from the fuel. Accordingly, the NRC staff concludes that it is unlikely that additional PSFs would be identified as RSFs for source terms outside the core, and the supplementary evaluations described in PSAR section 3.10 and evaluated in section 3.10 of this SE and the RSFs for radionuclide retention are adequate for the PSAR. The potential need for additional safety-significant radionuclide retention PSFs will be evaluated at the OL application stage when design and safety information for non-fuel sources is available.

An additional aspect of the LMGS design that impacts the RSFs is the use of inherent reactivity feedback as one of the two independent shutdown mechanisms and identification of inherent reactivity feedback as an RSF relied on to maintain safety. Due to its importance to safety, the NRC staff reviewed how the effectiveness of this RSF, which is under the control heat generation FSF, depends on the presence of high temperature, effectively creating a competing safety priority to the control heat removal RSF. As a result, as the reactor becomes more effective in the heat removal RSF, it becomes less effective in inherent reactivity and the control of heat generation RSF. Because of these competing RSFs, biasing parameters to the limiting value in one direction will not necessarily achieve a conservative result and the NRC staff discussed with the applicant that a more extensive transient analysis is needed to ensure that the range of potential conditions, including the impacts of uncertainties, have been evaluated.

Results in PSAR section 3.6 show the depressurized loss of forced cooling (DLOFC) event as bounding with respect to peak fuel temperature. The DLOFC event minimizes heat removal but would have stronger negative reactivity feedback as a result, when compared with other DBAs. To support confirming minimization of heat removal would result in bounding peak fuel temperature, considering the competing RSFs, the NRC staff reviewed a preliminary transient analysis in the regulatory audit. The preliminary transient analyses performed by LME, supported LME's position that the bounding fuel temperatures will occur when heat removal is most limiting. LME will evaluate the full range of key reactivity and thermal-hydraulic parameters across a range of cooling conditions to ensure that the events analyzed in the FSAR represent the worst-case or 95th percentile with adequate margin maintained for both safety functions. PSAR section 3.1.7 describes the analysis that will be completed to support the OL application.

PSAR table 5.2-1 lists the RFDCs corresponding to the RSFs. The RFDCs are evaluated in section 5.3 and chapters 6 and 7 of the SE.

As discussed in section 5.3 of this SE, the RFDCs were limited in LBE scope similar to the PDCs, creating inconsistency between the RFDCs and the LBEs where these functions are relied on to be performed within the PRA. More specifically, LME indicated that it limited the scope of the LBEs used to determine the RFDCs to the events that lead to the associated classification of the function. For example, LME indicated that only DBEs and DBAs lead to the identification of RSFs, based on the preliminary SSC classification methodology and the current set of LBEs not including any high-consequence BDBEs. Further, the NRC staff identified

multiple AOOs and BDBEs within the current LBE set with frequency bands extending into the DBE region that will be evaluated for RSFs through the updated LME methodology and are likely to be identified as additional LBEs driving classification of these functions. As discussed in section 5.1 of this SE, additional evaluations of frequency bands that extend into another LBE category were not performed in LME's preliminary SSC classification.

As explained in section 5.1 of this SE, the methodology has been updated to provide the evaluation of frequency bands that extend into another LBE category and is now acceptable. Accordingly, use of the updated methodology will achieve acceptable results. The potential safety gap between the design requirements and the safety analysis created by the omission of certain LBEs in the scope of the RFDCs is addressed by LME's consideration of the full scope of LBEs in the reliability and capability targets and special treatments to ensure the assumptions in the PRA are met by the design, as described in PSAR sections 6.3 and 7.2. Based on the approach described by LME, the PDCs and RFDCs should be expanded accordingly to avoid inconsistencies.

An example of the effect of the above limitation in RFDC scope is seen in the SGTL AOOs, SSGTL-01 and SSGTL-03, which both have 5th percentile frequencies extending into the DBE region. These AOOs should therefore be evaluated to identify any functions necessary to keep them below the F-C target and those functions should be identified as RSFs consistent with the NEI 18-04 methodology. From the preliminary set of consequence results provided in the PSAR, the NRC staff determined that closure of the steam generator isolation valves is a function that prevents these two AOOs from exceeding the F-C target, which should lead to a SR classification for this function and the identification of an RSF and application of the associated RFDC based on these AOOs. However, as stated in PSAR table 5.2-1, RSF 1.3, control water/steam ingress, and RSF 1.3.1, isolate water/steam sources, are limited to DBEs and DBAs, which is inconsistent with the LBEs driving classification since certain AOOs and BDBEs are not considered.

As determined in section 5.1 of this SE, it is unlikely that new RSFs/RFDCs will be identified as a result of additional evaluation. Additionally, in PSAR section 6.3, LME states that reliability and capability targets and special treatments for SR SSCs will cover the full scope of LBEs where the function is assumed to succeed within the PRA. The NRC staff will review the consistency between the LBEs driving classification and the RFDC scope at the OL stage based on the final set of LBEs and the final RFDCs.

Based on the foregoing, the NRC staff finds that the RSFs for the LMGS design at the CP stage are reasonable and consistent with guidance in RG 1.233 and RG 1.253. The NRC staff finds that the RSFs identified based on the preliminary design and safety analysis are acceptable to maintain the consequence of DBEs and AOOs and BDBEs that extend into the DBE range inside the F-C target, preventing high-consequence BDBEs from increasing beyond the F-C target, and conservatively ensuring that the 10 CFR 50.34 dose requirements are met in DBA analyses.

NSRST PSFs

The NRC staff reviewed PSAR section 5.2.2 to determine if the NSRST PSFs are reasonable and consistent with the guidance in RG 1.233 and RG 1.253. NSRST PSFs may be either risk-significant or required for DID adequacy depending on the SSC classification evaluation through which they were identified as NSRST. Section 5.1 of this SE provides the NRC staff's evaluation

of LME's SSC classification methodology. PSAR table 5.2-2 provides the list of NSRST PSFs and the associated CDCs for the preliminary LMGS design.

As discussed in section 5.1 of this SE, the evaluations used for identifying risk-significant NSRST functions were not performed during the preliminary SSC classification. As a result, some NSRST PSFs that are risk-significant are listed as NSRST for DID adequacy in the PSAR. The impacted systems are listed in RCI-1, confirmed by LME in the response to RCI-1 dated August 28, 2026 (ML26239A157).

The PSAR identified the following NSRST PSFs as required for DID adequacy, but through the preliminary evaluations documented in RCI-1, it appears that at least one subfunction in all four may be risk-significant[1]:

- PSF 1.1.3, "Control reactivity with movable poisons," as an NSRST PSF providing additional protection to RSF 1.1 which relies on inherent reactivity feedback;
- PSF 1.2.2, "Maintain core heat removal through active means," to provide additional protection to RSF 1.2; and
- PSF 1.3.2 and PSF 1.3.3, to provide additional protection to RSF 1.3, which isolates water/steam sources, by removing the water/steam source and preventing the occurrence of water/steam ingress events.

Additional PSFs not identified through the PRA but determined to be risk-significant in the preliminary SSC classification by the IDPP included:

- PSF 2, "Retain radionuclides in the HPB";
    - PSF 2.2, "Isolate HPB"; and
    - PSF 2.7, "Maintain the HPB pressure integrity."

These NSRST PSFs provide additional protection to the radionuclide retention aspect of RSF 1.

Similar to the approach with RFDCs discussed above, LME limited the LBE scope of CDCs based on limitations in their preliminary SSC classification methodology. For CDCs, LME's preliminary methodology only considered AOOs in identifying the LBEs driving classification rather than all LBEs, as indicated in the NEI 18-04 methodology. In its preliminary evaluations, LME limited many CDCs to AOOs. Using the updated methodology described in section 5.1 above, all LBEs will be evaluated when identifying NSRST SSCs, creating inconsistencies between the LBEs driving classification and the current scope defined in the CDCs. The safety impacts of this are being addressed in the same way as for the RFDCs, through ensuring consistency between the full LBE set that assumes success of these functions in the PRA and the reliability and capability targets and special treatments applied to the SSCs.

LME updated its SSC classification methodology as described in section 5.1 of this SE, which will ensure that the full scope of LBEs is evaluated for in the identification of NSRST PSFs. The

[1] Because LME performed the preliminary evaluation by failing SSCs and many SSCs perform multiple safety functions, the exact functions or subfunctions that are risk-significant were not identified.

NRC staff further determined, based on confirmation from LME as stated in PSAR section 7.2, that reliability and capability targets and special treatments for NSRST SSCs will cover the full scope of LBEs where the function is assumed to succeed within the PRA. Accordingly, the potential safety concern associated with the CDC scope is addressed. The NRC staff will review the consistency of LBEs driving classification and CDC scope at the OL stage based on the final set of LBEs and the final CDCs.

In summary, the NRC staff finds that the NSRST PSFs for the LMGS design at the CP stage are reasonable and consistent with the preliminary design and LME's preliminary SSC classification methodology. As discussed in this evaluation section and section 5.1 of this SE, corrections to the SSC classification methodology have been made to address differences between the preliminary LME methodology and NEI 18-04, and the resulting impact on the scope of safety-significant PSFs and SSCs is being incorporated into the safety analysis and design such that the results of the corrected methodology will be acceptable. The NRC staff finds that the content of application in PSAR section 5.2 conforms to the guidance in RG 1.253; specifically, the NRC staff finds that the applicant identified the NSRST PSFs that are either risk-significant or required for DID adequacy, consistent with the RG 1.253 criteria.

5.2.2 Conclusion

Based on the foregoing, the NRC staff finds that the safety-significant PSFs (i.e., RSFs and NSRST PSFs) described in PSAR section 5.2 are reasonable and sufficient to meet the regulatory requirements in 10 CFR 50.34(a)(1) and 10 CFR 50.34(a)(4) for the issuance of a CP in accordance with 10 CFR 50.35 based on the preliminary PRA and design. The NRC staff will reevaluate the safety-significant PSFs against the final PRA and the final design during the review of the FSAR as part of a future OL application.

5.3 Principal Design Criteria

PSAR section 5.3 describes the LMGS PDCs. PSAR section 5.3 states that the NRC-approved TR, "Xe-100 Principal Design Criteria Licensing Topical Report," Revision 3 (ML24319A155) is IBR, in part, and provides the departures from the TR for LMGS.

The applicable regulatory requirements for the evaluation of PDCs are as follows:

- 10 CFR 50.34(a)(3)(i).

The applicable guidance for the selection of PDCs is as follows:

- DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications – Roadmap"; Section 1.1.4;
- RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors." (ML17325A611);
- RG 1.233, Revision 0, which endorses NEI 18-04, Revision 1, with clarifications; and
- RG 1.253, Revision 0, which endorses NEI 21-07, Revision 1, with clarifications and additions.

5.3.1 Technical Evaluation

The NRC staff reviewed PSAR section 5.3 using the guidance discussed above to determine if the PDCs are acceptable. The NRC staff's SE (ML24284A012) for the Xe-100 PDC TR imposed three limitations and conditions (L&Cs) that must be addressed by applicants when incorporating the TR by reference. The NRC staff's evaluation of how the Xe-100 PDC TR L&Cs are met for LMGS is presented below. In addition, the NRC staff's evaluation of the LMGS departures from the Xe-100 PDC TR is included in the evaluation of limitation and condition 1 below.

Limitations and Conditions

In its SE on the Xe-100 PDC TR, the NRC staff included the following conditions:

Condition (1): X-energy is requesting approval for the proposed PDCs based on a preliminary design and the LMP implementation at the time of the TR submittal. Xe-100 design changes and associated LMP implementation could necessitate a revision to the proposed PDCs described in the TR. Therefore, future licensing applicants referencing the TR must confirm that the PDCs in this TR remain appropriate for its design. In addition, if additional or revised PDCs are identified that are not within the scope of what is approved in this TR, those PDCs will be subject to further NRC staff review.

In its review of the LME CP application, the NRC staff also examined the relevant design information discussed in the PSAR to determine if the PDCs are appropriate and will function as designed, consistent with the analyses in the PSAR.

Limited LBE Scope

In section 2.2.2 of the SE for the Xe-100 PDC TR, the NRC staff noted that BDBEs were absent from the PDC scope and conditioned the NRC staff's acceptance of the scope of LBEs in the TR on future license applicants confirming that no safety-significant SSCs associated with BDBEs exist based on the design described in the license application and implementation of the LMP process. Additionally, condition 1 of the SE highlights the preliminary nature of the design and that the PDCs as accepted in the Xe-100 PDC TR may not remain appropriate as the design matures and LMP is implemented. During the review of the LMGS CP application, the NRC staff identified that limiting the scope of PDC-RFDC to DBE/DBAs and PDC-CDC to AOOs is inconsistent with NEI 18-04 and NEI 21-07, as endorsed by RG 1.233 and RG 1.253, respectively. This narrowing of PDC scope also results in inconsistencies between the design requirements imposed by the PDCs on the safety-significant SSCs and the full scope of LBEs for which they are assumed to function in the PRA, as discussed in section 5.2 of this SE. Further, PSAR sections 5.4 and 5.5, which tabulate LBEs for which SR and NSRST SSCs perform prevention and mitigation safety functions, include LBEs outside the scope of the applied PDCs.

The NRC staff's review finds that the PDC LBE scope limitation is addressed by complementing the PDCs by an iterative evaluation and design process. This process was explained in the regulatory audit and is summarized in PSAR sections 6.3 and 7.2. The SSC performance is evaluated through this process for all LBEs credited with performing prevention and mitigation safety functions, and that evaluation feeds back into the design through SSC-specific design requirements, reliability and capability targets, and special treatments. The NRC staff

determined that this approach ensures the assumptions in the PRA are met by the design. Because the reliability and capability targets and special treatments from this process are necessary to address this gap, the NRC staff considers these principal architectural and engineering criteria. In reaching its decision concerning the reasonable assurance, the NRC staff was guided by, among other things, considerations of how the processes described in the PSAR will address any gaps in the PDCs. Based on the evaluation described above, the NRC staff finds the PDC LBE scope limitation acceptable at the CP stage and the NRC staff will evaluate PDCs for consistency with the final set of LBEs at the OL stage.

PDCs with Revised Criterion and Basis

The criteria and bases for PDCs 19, 20, 26, 70, and 71 that differ from the approved Xe-100 PDC TR as described in PSAR section 5.3 are evaluated below:

PDC 19: During the audit, it was noted that PDC 19 in the approved PDC TR was no longer adequate because an NSRST PSF that relies on operator action in the control room was identified. As a result, LME submitted a new PDC-CDC 19 to supplement the existing PDC 19 from the approved PDC TR. The NRC staff reviewed the new PDC-CDC 19 and associated basis. The NRC staff determined that this criterion is the same as Xe-100 PDC 19 criterion found in the Xe-100 PDC TR with the exception that PDC-CDC 19 contained modified language aligning the functional scope to LBEs for which control room actions are needed to perform prevention and mitigation safety functions. The basis was updated in the PSAR to reflect a revised PRA which identifies additional LBEs with operators performing an NSRST PSF. Based on the review discussed above, the NRC staff finds the new PDC-CDC 19 and the revised basis to be acceptable and consistent with PRA iteration, design progression, and implementation of the LMP process described in NEI 18-04 and endorsed by RG 1.233.

PDC 20: The NRC staff reviewed PDC 20 and the associated basis. The NRC staff determined that the criterion is consistent with the approved Xe-100 PDC TR except for being identified as an RFDC due to its functional basis. The basis was updated in the PSAR to include alignment with applicable RSFs. Because the designation of PDC 20 as an RFDC does not limit it compared to PDC 20 approved in the TR, the NRC staff finds that these departures from the PDC TR are acceptable and consistent with PRA iteration and implementation of the LMP process described in NEI 18-04 and endorsed by RG 1.233.

PDC-RFDC 26 and PDC-CDC 26: The NRC staff reviewed the PDC-CDC 26 criterion, as well as the bases for PDC-RFDC 26 and PDC-CDC 26. This criterion clarifies that the scope of PDC 26 applies to movable poisons, while the other means of reactivity control is provided by PDC 11. Additionally, the bases reflect the continuing PRA iteration and alignment with RSFs and PSFs. The NRC staff finds these revisions to be acceptable because PDC 26 along with PDC 11 addresses the underlying purpose of MHTGR-DC 26 in RG 1.232 by setting performance goals for providing adequate control over reactivity, and the bases are consistent with PRA iteration and implementation of the LMP process described in NEI 18-04 and endorsed by RG 1.233.

PDC-RFDC 70: The NRC staff reviewed the departures from the version of PDC-RFDC 70 approved in the TR and the associated bases, which replaced “helium pressure boundary and core internal structures” with “reactor systems capable of affecting core geometry” among other revisions to align with the associated RSF. The NRC staff finds the departures acceptable because PDC-RFDC 70 reflects the LMGS design, while still meeting the underlying purposes

for the associated RSFs of pertinent portions of MHTGR-DCs 14 and 70 in RG 1.232, by setting performance goals for maintaining core geometry.

PDC-RFDC 71: The NRC staff reviewed the departures from the version of PDC-RFDC 71 approved in the TR, and associated bases. The NRC staff finds the updates to PDC-RFDC 71 and the associated basis to be acceptable because they reflect PRA iteration and alignment with RSFs and are consistent with the LMP process described in NEI 18-04 and endorsed by RG 1.233, specifically, “reactor building” was replaced with “reactor support system and structure” to more accurately identify the SSCs needed to structurally protect the reactor vessel and reactor system geometry.

PDCs with Revised Bases

The bases for PDCs 10, 13, 14, 15, 16, 17, 30, 31, 32, 34, and 44 that differ from the bases in the approved Xe-100 PDC TR are evaluated below:

PDC 14 and PDC 31: The NRC staff reviewed the revised bases for PDCs 14 and 31. The NRC staff finds the revised bases acceptable because the criteria for PDCs 14 and 31 remain unchanged and the revised bases clarify that fracture prevention of the HPB in general is addressed with PDC 31. The NRC staff also determined that rapidly propagating failure of the portions of the HPB that affect core geometry is adequately addressed in PDC-RFDC 70.

PDC 32: The NRC staff reviewed the revised basis for PDC 32 (replaced by PDC-6) and finds that the underlying substance is unchanged and is acceptable.

PDC-RFDC 34: The NRC staff reviewed the revised basis for PDC-RFDC 34, which provides additional clarification that the underlying purpose of “effective heat removal” is to maintain both the fuel and core structural component temperatures below acceptable thermal limits. The NRC staff finds this basis acceptable because it is adequate to ensure that the reactor cavity cooling system be designed to provide heat removal for both the fuel and core structural components needed to maintain core geometry.

Updated Bases for PDCs to Align with Revised RSFs and PSFs

The NRC staff reviewed updated bases for several PDCs not discussed above, including PDC 10, PDC-RFDC 13, PDC-CDC 13, PDC-CDC 15, PDC-RFDC 16, PDC-CDC 16, PDC-CDC 17, PDC-RFDC 30, PDC-CDC 30, PDC-CDC 34, and PDC-CDC 44, to reflect the continuing PRA development and LMP implementation and alignment of RSFs and PSFs. The NRC staff finds the updates to these bases to be acceptable because they do not reduce the performance goals for the listed PDC set in the TR and are consistent with PRA iteration and implementation of the LMP process described in NEI 18-04 and endorsed by RG 1.233.

Condition (2): For PDC-RFDC 11, X-energy used the words “in the power operating range.” This phrase has the potential to complicate the applicability of PDC-RFDC 11 to the scope of MHTGR-DC 12. Power oscillations or reactivity upsets that may occur outside the power operating range should be assessed. Accordingly, the NRC staff conditions the acceptance of PDC-RFDC 11 on confirmation by future licensing applicants referencing this TR that applicable safety analyses cover the full scope of the operating range, DBE, and DBA conditions for the final design.

For condition 2, LME stated in PSAR table 1.4-7, that the condition will be met via safety analyses that cover the full scope of the operating range, DBE, and DBA conditions for the final design as part of the OL application. The NRC staff finds this approach acceptable because condition 2 is for the final design.

Condition (3): For PDC 14, X-energy is proposing to delete the words “ingress of air, secondary coolant, or other fluids” from MHTGR-DC 14 and replace it with the words “moisture ingress.” For justification, X-energy states that “no risk-significant AOOs, DBEs, or DBAs were identified with unacceptable ingress of air or other fluids.” This cannot be verified at the present state of the Xe-100 design. Accordingly, the NRC staff conditions the acceptance of this PDC on the NRC staff’s review of AOOs, DBEs, and DBAs as part of a future license application referencing this TR.

For condition 3, the NRC staff reviewed the AOOs, DBEs, and DBAs provided in PSAR chapter 3 and did not identify risk-significant LBEs with unacceptable ingress of air or other fluids. Therefore, the NRC staff finds that condition 3 is met for the CP stage and will confirm at the OL review stage that there are no risk-significant LBEs with unacceptable ingress of air or other fluids.

In summary, the applicant IBR, with some departures, the NRC-approved PDC TR. The NRC staff reviewed how the applicant addressed the L&Cs and the departures from the TR as discussed above. The NRC staff determined that the departures reflect the updated design and analysis and are acceptable when evaluated using the applicable guidance discussed earlier. Thus, the NRC staff finds that the applicant provided a set of PDCs, complemented by a robust design process and reliability and capability targets for safety-significant SSCs through implementation of LMP, that are appropriate for establishing requirements for the LMGS design and establish the necessary design, fabrication, construction, testing, and performance design criteria for safety-significant SSCs to provide reasonable assurance that the LMGS design could be operated without undue risk to the health and safety of the public. The NRC staff expects that PDCs may need to be revised as the design progresses and the LMP methodology is implemented.

5.3.2 Conclusion

Based on its review, the NRC staff finds that the information on PDCs, described in PSAR section 5.3, meets the regulatory requirements in 10 CFR 50.34.

5.4 Safety-Related SSCs

This section of the SE documents the NRC staff’s review of PSAR section 5.4 which identifies the SR SSCs selected to perform the RSFs. PSAR table 5.4-1 through table 5.4-11, present the SR SSCs, the preliminary list of LBEs (AOOs, DBEs, and BDBEs) where those SSCs perform their function, and the RSFs responsible for preventing or mitigating each of these LBEs. The NSRST SSCs are addressed in PSAR section 5.5.

The applicable regulatory requirements for the evaluation of SR SSCs are as follows:

- 10 CFR 50.34(a)(1);

- 10 CFR 50.34(a)(4); and
- 10 CFR 50.35.

The applicable guidance for the selection of SR SSCs is as follows:

- RG 1.233, Revision 0, which endorses NEI 18-04, Revision 1, with clarifications.
- RG 1.253, Revision 0, which endorses NEI 21-07, Revision 1, with clarifications and additions.

5.4.1 Technical Evaluation

The NRC staff reviewed PSAR section 5.4 to determine if the list of SR SSCs is consistent with the regulatory guidance in RG 1.233 and RG 1.253 and the SSC classification methodology described in PSAR section 5.1. The NRC staff's evaluation of the SSC classification methodology is documented in section 5.1 of this SE. As part of the evaluation of PSAR section 5.4, the NRC staff audited relevant design documents to assess the roles that the SSCs play in ensuring safety, from a frequency, consequence, and DID perspective and to confirm that the results are accurately captured in the PSAR.

PSAR section 5.1 states that the applicant follows the risk-informed and performance-based methodology in NEI 18-04, endorsed in RG 1.233, to classify SSCs. Based on the methodology, the applicant identified the SR SSCs of the LMGS design in PSAR tables 5.4-1 through 5.4-11 as follows:

- Uranium pebbles
- Reactor protection system
- Nuclear instrumentation system
- Graphite core assembly
- Core metallic structures
- Reactor control & shutdown system
- Reactor pressure vessel
- Reactor cavity cooling system
- Steam generator system
- Primary pressure relief system

These 10 SR SSCs are those modeled in the LMGS PRA as discussed in section 3.1.1 of the SE. LME identified four SR SSCs that were not modeled in the PRA for the preliminary design. These SSCs are not included in the PSAR section 5.4 tables, and they are:

- Reactor building
- Defuel chute
- Defuel chute pressure vessel
- Reactor vessel support system

PSAR section 6.4 describes design aspects of the SR SSCs to ensure that they can perform the RSFs relied on within the safety analysis. The NRC staff evaluates these aspects of the SR SSCs in chapter 6 of this SE. Accordingly, the NRC staff's evaluation of this section is focused

on the information in PSAR tables 5.4-1 through 5.4-11, which includes the SR SSCs modeled in the PRA, the preliminary list of associated LBEs, and the RSFs responsible for preventing or mitigating each of the LBEs.

For the SSCs not modeled in the PRA, the reactor building and reactor vessel support system are evaluated as part of the reactor pressure vessel, as they are essential support for the reactor pressure vessel and the reactor system. The defuel chute provides essential support for both the fuel and the defuel chute pressure vessel, ensuring heat removal to prevent high temperature in the fuel from challenging the radionuclide retention and high temperatures in the defuel chute pressure vessel that would potentially challenge its function of maintaining core geometry.

The NRC staff has determined that the information on the SR SSCs in PSAR section 5.4 is consistent with the guidance in NEI 21-07 as endorsed with clarifications by RG 1.253. Regarding consistency with RG 1.233, the NRC staff determined that the inconsistencies in applying the NEI 18-04 process did not have a significant impact on the identification of SR functions or the selection of SR SSCs to perform those functions. This determination is based on LME performing an evaluation in which all non-SR SSCs are modeled as failed and frequencies and consequences of all AOOs and DBEs were confirmed to be below the F-C target at the 95th percentile. While LME's use of the 95th percentile versus the mean and modeling all non-SR functions as failing at once in its evaluation is conservative, LME did not evaluate BDBEs with frequency bands extending into the DBE region. The identification of SR SSCs in NEI 18-04 is based on comparing the mean to the F-C target and the mean values are generally significantly below the 95th percentile values for the LBEs in the PSAR. As a result, the NRC staff determined that it is unlikely that additional SR SSCs will be identified from these events. As documented in RCI-1 (ML26230A155) and confirmed by LME in its response to RCI-1, a preliminary evaluation of BDBEs was performed against the F-C target. The NRC staff confirmed in the regulatory audit that, as anticipated, no additional SR SSCs were identified during this preliminary BDBE evaluation.

Table 5.4-1 of this SE summarizes the high-level RSFs that each SR SSC contributes to achieving. LME identified SSCs not modeled in the PRA as SR based primarily on their roles in maintaining reactor support geometry (RSF 1.4.2) and maintaining core geometry (RSF 1.4.1). Based on the processes used to identify SR SSCs, which included LBE evaluations from the PRA, DBA analysis, insights from the IDPP, and a comparison of the associated LBE set with the functions of the SR SSCs, the NRC staff finds the applicant's preliminary list of SR SSCs to be reasonable and sufficient for meeting requirements of 10 CFR 50.34(a)(1), which requires the applicant to include in the application, a description and safety assessment of the facility, and 10 CFR 50.34(a)(4), which requires a preliminary analysis and evaluation of the design and performance of SSCs of the facility with the objective of assessing the risk to public health and safety.

The NRC staff notes that the SR SSCs identified in this PSAR section are based on the preliminary design and safety analysis available at the time LME filed the CP application and may change at the OL stage. The risk-informed and performance-based methodology in RG 1.233 facilitates a systematic iterative process for design development, PRA development, selection of LBEs, classification of SSCs, and evaluation of DID adequacy. Therefore, the NRC staff will review the final list of the SR SSCs submitted in the OL application, when the LMP process, consistent with the final design, is complete.

5.4.2 Conclusion

Based on its review, as described above, the NRC staff finds that the information on SR SSCs described in PSAR section 5.4 is consistent with RG 1.233 and RG 1.253, and meets the regulatory requirements in 10 CFR 50.34(a)(1) and 10 CFR 50.34(a)(4) for the issuance of a CP in accordance with 10 CFR 50.35.

Further technical or design information required to finalize the identification of SR SSCs and associated RSFs as part of the NEI 18-04 process implementation can reasonably be left for later consideration. The NRC staff will confirm that the final identification of SR SSCs conforms to the relevant guidance listed above during the evaluation of the FSAR as part of the OL application review.

Table 5.4-1: Summary of SR SSCs and Associated High-Level RSFs

SR SSC	RSF 1, retain radionuclides in fuel particles and pebbles	RSF 1.1, control reactivity	RSF 1.2, control heat removal	RSF 1.3, control water/steal ingress	RSF 1.4, maintain geometry
Uranium pebbles	X	X	X		
Reactor protection system		X		X	
Nuclear instrumentation system		X		X	
Graphite core assembly		X	X		X
Core metallic structures		X	X		X
Reactor control & shutdown system		X			
Reactor pressure vessel			X		X
Reactor cavity cooling system (passive portion)			X		
Steam generator system (isolation valves)				X	
Primary pressure relief system					X
Defuel chute pressure vessel			X		X
Reactor building (reactor support structure and shared foundation)					X
Defuel chute			X		X
Reactor vessel support system					X

5.5 Non-Safety-Related SSCs with Special Treatments

PSAR section 5.5 describes the NSRST SSCs. The description of the SSC classification process is addressed in PSAR section 5.1, and SR SSCs are addressed in PSAR section 5.4.

The applicable regulatory requirements for the evaluation of NSRST SSCs are as follows:

- 10 CFR 50.34(a)(1); and
- 10 CFR 50.34(a)(4).

The applicable guidance for the selection of NSRST SSCs is as follows:

- RG 1.233, Revision 0, which endorses NEI 18-04, Revision 1, with clarifications.
- RG 1.253, Revision 0, which endorses NEI 21-07, Revision 1, with clarifications and additions.

5.5.1 Technical Evaluation

The NRC staff reviewed PSAR section 5.5 to determine if the list of NSRST SSCs is consistent with the regulatory guidance in RG 1.233 and RG 1.253 and the SSC classification methodology described in PSAR section 5.1. The NRC staff's evaluation of the SSC classification methodology is documented in section 5.1 of this SE. As part of the evaluation of PSAR section 5.5, the NRC staff audited relevant design documents to assess the roles the SSCs play in ensuring safety from a frequency, consequence, and DID perspective and to confirm that the results are accurately captured in the PSAR.

PSAR section 5.1 states that the applicant follows the risk-informed and performance-based methodology in NEI 18-04, endorsed in RG 1.233, to classify SSCs. The applicant classifies non-safety-related SSC as NSRST if they either perform risk-significant functions or perform functions that are necessary for DID adequacy. For the former, the applicant classified SSCs as risk-significant if the SSC function is necessary to keep any LBEs inside the F-C target at the 95^{th} percentile frequency and consequence or if the total frequency of LBEs with the SSCs modeled as failed is within 1 percent of any of the three cumulative risk targets based on mean frequencies and consequences.

In PSAR section 5.5.1, the applicant identified four NSRST SSCs that perform risk-significant functions based on the total frequency of LBEs with the SSCs modeled as failed being within 1 percent of one or more of the three cumulative risk targets. This is consistent with the guidance in NEI 21-07, Section 5.5.1. The NSRST SSCs are as follows:

- Fuel handling system
- Steam generator system
- Investment protection system
- Nuclear instrumentation system

PSAR table 5.5-22 shows that all these systems perform the PSF of isolating systems connected to the HPB. The nuclear instrumentation system is SR; however, it is listed here because the nuclear instrumentation system performs both RSFs and NSRST PSFs.

In PSAR section 5.5.2, the applicant identified the following systems as performing functions necessary for DID adequacy. As discussed in section 5.2 of this SE, a preliminary risk-significance evaluation, the results of which are described in RCI-1 (ML26230A155) and confirmed by LME in the response to RCI-1, impacted the risk category of some of the NSRST SSCs. The SSCs that perform functions for DID adequacy, incorporating information from RCI-1, are as follows:

- Distributed control system
- Hot gas duct
- Nuclear island heating, ventilation, and air conditioning system
- Reactor cavity cooling system, active portion
- Cross vessel, HPB
- Defuel chute pressure vessel, HPB
- Fuel handling system, HPB
- Helium circulator system, HPB
- Helium service system, HPB
- Primary pressure relief system, HPB
- RCSS pressure vessel, HPB
- Reactor pressure vessel, HPB
- Steam generator system, HPB

The SSCs that perform functions identified in the PSAR as NSRST for DID that are likely to shift to risk-significant based on the results reflected in RCI-1 are as follows:

- Distributed control system
- Reactivity control and shutdown system
- Investment protection system
- Reactor building cooling water system
- Nuclear instrumentation system
- Nuclear island electrical system
- Helium circulator system
- Start-up and shutdown system
- Steam generator system
- Shield structure, reactor building pressure relief

Some of the systems in both lists are SR and are listed here in addition to section 5.4 of this SE because the SR SSCs perform both RSFs and NSRST PSFs. Table 5.5-1 of this SE provides a summary table of the NSRST SSCs showing the associated high-level NSRST PSFs and the RSFs protected by the NSRST PSF.

The list in RCI-1 also includes one system classified as NST in the PSAR. This is the conventional island electrical system. The conventional island electrical system provides electrical power to the nuclear island electrical system from the utility electric grid and the turbine generator. As described in PSAR section 7.3.8, the conventional island electrical system

provides power to the nuclear island electrical system during normal operation, but if power is lost, the nuclear island electrical system automatically switches to the standby diesel generators (SDGs). The SDGs are the primary electrical source relied on for performing NSRST functions. With six SDGs and with further development of the PRA and reduction of the uncertainties in the consequence evaluations between the CP and OL, the NRC staff determined that there is a reasonable likelihood that the conventional island electrical system will be NST at the OL stage, consistent with what is currently reflected in the PSAR. As determined in section 5.1 of this SE, the SSC classification methodology provides confidence that the final risk-significance determinations will be satisfactory.

In PSAR table 5.5-21, the applicant's IDPP also identified NSRST functions related to SSCs not modeled in the PRA. These NSRST SSCs are associated with preventing challenges to other SSCs that perform safety-significant PSFs and are as follows:

- Fuel handling system
- Steam generator support structure
- Shield structure
- Fuel handling annex building
- Inter unit access tunnel
- Spent fuel intermediate storage facility
- Canister processing facility
- Nuclear island auxiliary structure
- Controls and electrical building
- Spent fuel racks
- Spent fuel canisters

Some systems may show up in multiple lists because they perform multiple safety-significant functions that are identified through different assessments within the SSC classification process.

In PSAR table 5.5-23, the applicant identified a safety-significant operator action that supports NSRST PSF 1.2.2.2.4, maintain active core heat removal during events that involve water/steam ingress. This operator action involves identifying the affected bundle in the steam generator, isolating it, and restoring active cooling to keep fuel temperatures low and mitigate the radionuclide release.

The NRC staff notes that the NSRST SSCs are described in PSAR section 7.3 or in PSAR section 6.4 (e.g., the shield structure in PSAR section 6.4.1.2 and the steam generator support structures in PSAR section 6.4.1.3), depending on whether they also perform an RSF. The NRC staff evaluated the descriptions and analyses of the NSRST SSCs in chapter 6 and 7 of this SE. Accordingly, the NRC staff's review of this section is focused on the information in this PSAR section, including results provided in RCI-1, which includes the NSRST SSCs modeled in the PRA, the LBEs, and the PSFs responsible for preventing or mitigating each of the LBEs.

The NRC staff has determined that the information on the NSRST SSCs in PSAR section 5.5 is consistent with the guidance in NEI 21-07 as endorsed with clarifications by RG 1.253. Regarding consistency with RG 1.233, as discussed in section 5.1 of this SE, the preliminary SSC classification methodology followed for the PSAR had some inconsistencies with the NEI 18-04 process. The inconsistencies are discussed in detail in section 5.1 of this SE, and the applicant has updated its methodology to ensure greater consistency with NEI 18-04. The

results of the updated methodology are reflected in RCI-1 but were not incorporated into the PSAR. The NRC staff determined that the inconsistencies in applying the NEI 18-04 process did not have a significant impact on the identification of NSRST functions or SSCs but did impact the risk-significance determinations. These impacts to the risk significance determinations are identified above and discussed in section 5.2 of this SE.

The NRC staff notes that the NSRST SSCs identified in this PSAR section are based on the preliminary design and safety analysis at this CP stage and may change at the OL stage. The risk-informed and performance-based methodology in RG 1.233 facilitates a systematic iterative process for design development, PRA development, selection of LBEs, classification of SSCs, and evaluation of DID adequacy. Therefore, the NRC staff will review the final list of the NSRST SSCs at the OL application stage when the LMP process, consistent with the final design, is complete.

Based on the foregoing, the NRC staff finds the preliminary list of NSRST SSCs in PSAR section 5.5 along with the additional information confirmed through RCI-1 to be reasonable and sufficient for meeting requirements of 10 CFR 50.34(a)(1), which requires an applicant to include in the application a description and safety assessment of the facility and 10 CFR 50.34(a)(4), which requires a preliminary analysis and evaluation of the design and performance of SSCs of the facility with the objective of assessing the risk to public health and safety. Additional information on the design and performance of these SSCs is located in PSAR sections 6.4 and 7.3 and evaluated in sections 6.4 and 7.3 of this SE, respectively.

5.5.2 Conclusion

Based on its review, as described above, the NRC staff finds that the information on the NSRST SSCs, described in PSAR section 5.5 and RCI-1, is consistent with RG 1.233 and 1.253 and meets the regulatory requirements in 10 CFR 50.34(a)(1) and 10 CFR 50.34(a)(4) for the issuance of a CP in accordance with 10 CFR 50.35.

The NRC staff will confirm that the final NSRST SSCs conforms to the relevant guidance listed above during the evaluation of the FSAR as part of the OL application review.

Table 5.5-1: Summary of NSRST SSCs and Associated High-Level PSFs

RSF Protected by NSRST PSF -->	**RSF 1.1**	**RSF 1.2**	**RSF 1.3**		**RSF 1**	
NSRST SSC	**PSF 1.1.3**	**PSF 1.2.2**	**PSF 1.3.2**	**PSF 1.3.3**	**PSF 2.2**	**PSF 2.7**
Reactivity control and shutdown system	X					X
Reactor building cooling water	X	X				
Core metallic structures	X	X				
Graphite core assembly	X	X				
Defuel chute		X				
Investment protection system	X		X		X	
Nuclear instrumentation system	X	X	X			
Nuclear island electrical	X	X	X		X	
Nuclear island HVAC	X	X	X		X	
Hot gas duct		X				
Helium circulator system		X				X
Reactor system		X				X
Steam generator system		X	X	X	X	X
Start-up and shutdown system		X	X			
Reactor cavity cooling system		X				
Distributed control system		X				
Helium purification system						X
Fuel handling system					X	X
Primary pressure relief system						X
Reactor pressure vessel						X
Nuclear island civil structures (shield structure, steam generator support structure, fuel handling auxiliary building, spent fuel intermediate storage facility, inter unit access tunnel, canister processing facility)	provide essential support for the SSCs within					

* Reactor system includes reactor pressure boundary system, reactor pressure vessel, defuel chute pressure vessel, cross vessel, reactor control and shutdown system pressure vessel, and primary pressure relief system.

6 SAFETY-RELATED STRUCTURE, SYSTEM, AND COMPONENT CRITERIA AND CAPABILITIES

This chapter of the safety evaluation (SE) summarizes the U.S. Nuclear Regulatory Commission (NRC or Commission) staff's review and evaluation of the Long Mott Generating Station (LMGS) preliminary safety analysis report (PSAR) chapter 6, "Safety-related SSC [Structures, Systems, and Components] Criteria and Capabilities."

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

PSAR section 6.1 describes the determination of 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 (SR) SSCs are as follows:

- 10 CFR 50.34(a)(3)(ii); and
- 10 CFR 50.34(a)(4).

The applicable guidance for the review of the design requirements for SR SSCs are as follows:

- Regulatory Guide (RG) 1.76, Revision 1, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," (ML070360253).
- RG 1.115, Revision 2, "Protection Against Turbine Missiles," (ML101650675).
- RG 1.189, Revision 5, "Fire Protection for Nuclear Power Plants," (ML23214A287).
- RG 1.221, Revision 0, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," (ML110940300).
- RG 1.233, Revision 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, Nuclear Energy Institute (NEI) 18-04, Revision 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," (ML19241A336), with clarifications.
- RG 1.247, Revision 0, "Acceptability of Probabilistic Risk Assessment Results for Non-Light-Water Reactor Risk-Informed Activities," For Trial Use (ML21235A008).
- RG 1.253, Revision 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 NEI 21-07, Revision 1, "Technology Inclusive Guidance for Non-Light Water

Reactor Safety Analysis Report: For Applicants Utilizing NEI 18-04 Methodology," (ML 22060A190), with clarifications and additions.

- Division of Advanced Reactors and Non-Power Production and Utilization Facilities (DANU) Interim Staff Guidance (ISG) 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 U.S. Nuclear Regulatory Commission (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, respectively, which endorse 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 Title 10 of the *Code of Federal Regulations* (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's review determined that Long Mott Energy, LLC., (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 principal design criteria (PDC) 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, high-energy line break (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 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants." 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 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 (PRA) 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 that 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, Revision 1 and RG 1.221. The NRC staff determined, in sections 2.3 and 3.7 of this SE, that the site-specific wind hazard parameters and associated applicant's assessment are consistent with RG 1.76, Revision 1 and RG 1.221. Further, the NRC 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 NRC staff's determinations documented in sections 2.3 and 3.7 of this SE, the NRC staff finds that the wind hazard DBHL is consistent with 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 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, National Fire Protection Association (NFPA) 804, and the non-light-water reactor (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

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 (93°C) or a maximum operating pressure exceeding 275 psig (1.90 MPag). 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 NRC 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's 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 2.2 of this SE and finds that LME has shown that hazards from nearby external facilities and transportation routes are not expected to develop into a design-basis hazard. The NRC 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 its associated DBHLs as necessary and applicable. The NRC 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's 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 is therefore 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 non-safety-related with special treatment (NSRST) and non-safety-related with no special treatment (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 (RGs). The NRC staff's review finds 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, PRA safety functions (PSFs) 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's 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 NRC staff's evaluation of the seismic design response spectra (DRS) 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's 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's 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's 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. 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 NRC 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 NRC 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 quality assurance (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 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); and
- 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," Revision 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," Revision 1 (ML19241A472).
- RG 1.253, Revision 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," Revision 1 (ML22060A190).

6.2.1 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 PSAR section 6.2.

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-design-basis accident (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 the NRC staff's findings pursuant to 10 CFR 50.34(a)(4) and 10 CFR 50.35(a)(1).

The preliminary targets provided in the PSAR are limited to those for RSFs performed by SR SSCs that are explicitly modeled in the 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 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 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 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 that 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 F-C 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.

As described in PSAR sections 6.3 and 7.2, LME has established a process for assigning reliability and capability targets and special treatments that ensures that the full scope of LBEs are addressed within the design of the SSCs. This process is necessary to address the gap that exists between the PDCs LME has established and the LBEs in which the SSCs are assumed to function within the safety analysis. Because the reliability and capability targets and special treatments from this process are necessary to address this gap, the NRC staff considers these principal architectural and engineering criteria. In reaching its decision concerning the reasonable assurance, the NRC staff was guided by, among other things, considerations of how the processes described in the PSAR will address any gaps in the PDCs.

6.2.2 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).

6.3 Special Treatment Requirements for Safety-Related SSCs

PSAR section 6.3 describes the applicant's approach to selecting special treatment requirements for SR 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 SSCs perform their design-basis functions."

The applicable regulatory requirements for the evaluation of SR SSCs are as follows:

- 10 CFR 50.34(a)(4); and
- 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," Revision 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," Revision 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, Revision 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," Revision 1 (ML22060A190).

6.3.1 Technical Evaluation

RG 1.253, which endorses, with certain clarifications and additions, NEI 21-07, Revision 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 SR SSC considers the RSFs and 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, complementary design criteria (CDCs), RFDCs, and SRDCs, as discussed in section 5.3 of this SE. Because the special treatments are necessary to address the gap caused by the limited scope, the NRC staff consider these principal architectural and engineering criteria. In reaching its decision concerning the reasonable assurance, the NRC staff was guided by, among other things, considerations of how the processes described in the PSAR will address any gaps in the PDCs, CDCs, RFDCs, and SRDCs.

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 PSAR chapter 8 (e.g., QA, 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 that the systems can perform the RSFs for which they are being relied on, 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 SR 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).

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 SR SSCs. For systems that include both SR and 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 RSFs and NSRST PSFs, lists associated PDCs, provides the system-specific SRDCs and 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); and
- 10 CFR 50.35(a).

The 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."

The 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

The 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's 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. The NRC staff notes that the discussions pertaining to PDC 5 below refer to “modules” or “units.” The language in the approved PDC 5 for the Xe-100 design refers to “units;” however, for LMGS, “modules” are equivalent to “units” for the purpose of evaluating PDC 5.

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 design-basis events (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 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 that 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, the NRC staff will confirm that the safety aspects are addressed appropriately through the final reliability and capability targets and special treatments are 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 (SG) support structures (SGSTs), each supporting a SG; and an NSRST shield structure (SST) surrounding, and structurally separate from, the steel frames.

The system-specific regulatory requirements are as follows:

- 10 CFR Part 50, Appendix S.

The 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 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, the NRC staff evaluated their safety significance, compliance with applicable PDCs, and safety functions.

6.4.1.1.1 Reactor Support Structures and Common Safety-Related Foundation

The NRC staff reviewed PSAR section 6.4.1.1, which describes the RXSTs and common SR 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 NRC 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/reactivity control and shutdown system (RCSS) that are required to be in close physical proximity to the RPV/RCSS.

The RXSTs support the performance of RSF 1.4.2, 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 NRC 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 is consistent with PDC 1. The principal codes and standards include American Institute of Steel Construction (AISC) N690, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” American Concrete Institute (ACI) 349, “Nuclear Safety-Related Concrete Structures—Code Requirements and Commentary,” American Society of Civil Engineers (ASCE) 4, “Seismic Analysis of Safety-Related Nuclear Structures,” and ASCE 43, “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities.” 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 NRC 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 QA treatment applicable to the RXSTs and common SR foundation. PSAR section 6.4.1 states that the QA program (QAP) 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 Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” and American Society of Mechanical Engineers (ASME) NQA-1, “Quality Assurance Requirements for Nuclear Facility Applications.” The full scope of the LMGS QA 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, QA requirements for installation, inspection, and testing of structural concrete, structural steel, and foundations. ASME NQA-1 is acceptable to the NRC staff as endorsed by RG 1.28. PSAR table 1.4-2 states full conformance with RG 1.28.

The NRC staff's evaluation of the LMGS QAPD is provided in the QAPD Topical Report (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's regulatory guidance and the QAP requirements described above, the NRC staff finds that LME's preliminary design bases for the RXSTs and common SR foundation are consistent with 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 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 the 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 QA, 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 the 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, is consistent with PDC 2, and meets the requirements of 10 CFR Part 50, Appendix S, 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 the 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 the 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 the 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 is consistent with 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 (AOO), DBE, and DBA

conditions with the effects of 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 are consistent with 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's 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 are consistent with PDC 2 and meet 10 CFR Part 50, Appendix S. The NRC staff will review the final design against the requirements of PDC 2 and 10 CFR Part 50, Appendix S 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 noncombustible, 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 finds 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 are consistent with PDC 3. This conclusion is based on LME's use of construction materials that are high-temperature-resistant, inherently noncombustible, 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. The 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 NRC staff's 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 the consideration of 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 (HPB). 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 NRC 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. LME's preliminary assessment indicates that the maximum operating temperature for the RXSTs remains below 250°F (121°C), 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 NRC staff via review of supporting information during the audit, the NRC staff finds LME's approach for addressing temperature effects at the CP stage acceptable. The NRC 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 are consistent with 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 the RXSTs and common SR foundation to ensure that the LMGS design is consistent with 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 is consistent with PDC 5 because the RXSTs 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 NRC 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 is 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 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 Nuclear Management and Resources Council (NUMARC) 93-01, Revision 4F, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," 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 NRC staff's evaluation of the maintenance program is in section 8.3 of this SE. The NRC staff's 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 NRC staff finds that LME's preliminary design of the RXSTs and common SR foundation is 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. LME's approach for meeting the requirements of PDC 6 and the NRC 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 the consideration of normal, severe environmental, extreme environmental, and abnormal loads and respective combinations of those loads in accordance with American National Standards Institute (ANSI) 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 non-seismic demand and seismic demands. On this basis, the NRC staff finds that the RXSTs and common SR foundation will provide adequate structural support to protect the reactor geometry and therefore is consistent with PDC-RFDC 71. The NRC staff will review the final design against PDC-RFDC 71 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, through the 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 RSFs 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 NRC 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 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 NRC 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 the IDPP. PSAR table 5.5-21 indicates that the PSF for the SST is to prevent challenges to SR SSCs that perform the following RSFs:

- RSF 1 – retain radionuclides in fuel particles and pebbles
- RSF 1.1 – control reactivity
- RSF 1.2 – control heat removal
- RSF 1.4 – maintain geometry

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 to maintain core geometry, and 1.4.2 to maintain reactor support geometry, 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 a HELB anywhere within the HPB 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, "Guide for Emulating Cast-in-Place Detailing for Seismic Design of Precast Concrete Structures," and other codes and standards listed in PSAR 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 **[[ ]]** 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 QA 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 QAP described in part III of the LMGS QAPD and in PSAR section 8.1. Additional special treatments related to QA 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 QA 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 are consistent with PDC 1 because its 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 QAP. 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 **[[**

], 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, **[[**

]], the NRC staff finds that LME's approach for protecting the SST and shared NSRST foundation is acceptable, is consistent with PDC 2, and meets the requirements of 10 CFR Part 50, Appendix S, 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 is consistent with PDC 2. 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 the 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 NRC staff finds that LME's approach for protecting the SST from the flooding hazard is acceptable and is consistent with 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 are consistent with 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 are consistent with PDC 2. The NRC staff will review the final design against PDC 2 and 10 CFR Part 50, Appendix S, at the OL application stage.

Fire Protection (PDC 3)

The NRC staff's 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 NRC staff's findings regarding LME's approach for meeting the requirements of PDC 3 for the RXST and common SR foundation, the NRC staff finds that the design bases for the SST and shared NSRST foundation are consistent with 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 the 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 (65°C) which is controlled by the design function of the 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's 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 NRC staff via review of supporting information during the audit (see the discussion under "Safety Functions" below), the NRC staff finds LME's approach for addressing temperature effects at the CP stage to be acceptable. The NRC staff will review LME's adherence to the provisions in ACI 349-13 in the final design at the OL stage.

Based on the NRC staff's evaluation described above, the NRC staff finds that LME's preliminary design bases for the SST and shared NSRST foundation are consistent with 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 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 finds that LME's design bases for the SST and shared NSRST foundation 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 SST and shared NSRST foundation. 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 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 finds that LME's preliminary design of the SST and shared NSRST foundation is consistent with 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 **[[**

]]. 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 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

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 SGs 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 **[[**

]], 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 QA 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 QAP described in part III of the LMGS QAPD and PSAR section 8.1. Additional special treatments related to QA 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 QA 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 are consistent with PDC 1 because its 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 QAP. 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, is consistent with PDC 2, and meets the requirements of 10 CFR Part 50, Appendix S, with respect to the consideration of the effects from earthquakes.

Regarding the consideration of the effects of tornadoes and hurricanes, the NRC 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 the 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's review finds these strategies to be consistent with those applicable to the SR RXSTs and on this basis, the NRC staff finds that the flood protection strategies for the SGSTs are acceptable and are consistent with 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 are consistent with PDC 2. The NRC staff will review the final design against PDC 2 and 10 CFR Part 50, Appendix S, at the OL application stage.

Fire Protection (PDC 3)

The NRC staff's 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 NRC staff finds that the SGSTs are consistent with 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 the 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 is 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 NRC staff's review of fire protection is addressed under the evaluation of PDC 3. Consistent with the NRC staff's evaluation of how the SGSTs design bases are consistent with 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 are consistent with 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 SG and related components.

Based on the independence of each SGST (i.e., an individual and separate SGST for each SG and related components), the NRC staff determined that the SGSTs are not shared between the reactor modules. On this basis, the NRC 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 NRC 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 **[[ ]]**. 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 SGSTs. 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 (TS) 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 limiting conditions for operation (LCOs) identified in part IV of the CP application

identifies the availability of nuclear island heating, ventilation, and air conditioning (NIHV) and 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 that the information provided meets the requirements of 10 CFR 50.34(a)(5). The NRC staff's finding is based on the variables, conditions, or items that are likely to become subjects of TS for the RB being identified. Consistent with 10 CFR 50.34(b)(6)(vi), the NRC staff will evaluate the proposed TS when detailed design information is available at the OL application stage.

6.4.1.2 Conclusion

Based on the NRC 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 descriptions of preliminary and panned analyses and associated safety margins for structures and foundations.
- The information provided in part IV of the CP application, identifying preliminary RB conditions that are subjects of TS.

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 describes the RSFs and PDCs associated with the fuel.
- PSAR section 6.4.2.2 describes the fuel loading during initial reactor start and operation.
- PSAR section 6.4.2.3 describes the Xe-100 UCO fuel qualification and references TR, "Xe-100 Topical Report – TRISO-X Pebble Fuel Qualification Methodology," Revision 3, June 2023 (ML23160A293) (hereafter referred to as FQ-TR) which was reviewed and

approved in the NRC staff's SE, subject to three limitations (the NRC staff's SE is included in the FQ-TR (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 (SFFPs) are described in PSAR section 1.1.4.

The system-specific regulatory requirements applicable to the fuel are as follows:

- 10 CFR 50.34(a)(8).

The 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-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 that 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 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 (IBR) 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 American Society of Testing and Materials (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 finds the scope of the specification and the values to be consistent with the preliminary design and analysis in the PSAR.

Based on this information, the NRC staff finds that the preliminary fuel design is consistent with 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 RPV, RB, fuel handling system (FHS), and the 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 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, the 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, the 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 the 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 includes 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. The 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 FHS. 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, the 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. The 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 NRC staff finds that the preliminary design includes monitoring, inspection, testing, and surveillance functions for fuel that are consistent with the requirements of PDC 6. The NRC 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 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 found that of the 38 AOOs, 34 AOOs have a resulting mean dose consequence below 0.1 mrem (0.001 mSv). 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 depressurized loss of forced cooling (DLOFC) event was added for SD 068. The NRC staff finds 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 10 CFR 50.35(a)(2), and the NRC staff has reasonable assurance that the applicant will be able to demonstrate that the requirements of PDC 10 are 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 that 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 (SLs) 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 LME's preliminary transient analysis supported its 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 its 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 NRC

staff finds that the preliminary results show that both the fuel and the moderator temperature coefficients are strongly negative.

Based on this information, the 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 or low-enriched uranium 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 the role of fuel in providing the principal barriers as part of the functional containment, is evaluated by the NRC staff in section 3.1.4 of this SE. Details relative to the fuel qualification and associated impact on the functional containment are evaluated below. Based on the information provided in PSAR section 3.1.4, PSAR section 6.4.2, and the information IBR, 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 finds 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 consistent with 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, the 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 finds that 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 LUPs 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, the 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. The NRC 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 SFFPs and LUPs. SFFPs contain no uranium or other fissionable material, are NST, and are described in PSAR section 1.1.4.1.2. LUPs are identical to the HUPs used in steady-state refueling, but contain uranium enriched to **[[ ]]** versus the HUP uranium, which is enriched to 15.5 percent. Initially the core is filled with SFFPs. Fuel loading then begins with LUPs in a **[[ ]]** 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 **[[ ]]** LUPs have been loaded into the core or when the core is approximately **[[ ]]**.

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. **[ ]** 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 the 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 the 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 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 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. The NRC staff compared these dates to the completion of construction date provided in part I of the CP application, "General and Financial Information," section 1.4, "Construction Schedule," and determined that the irradiation campaign is scheduled to be completed before the completion of the construction because the test campaign is scheduled to be completed in Fall 2030, and the completion of the construction is expected to be completed no later than 2033. The completion of the 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 (L&C's) 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 the fuel qualification and testing which requires R&D 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)	The NRC Staff's 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 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	The 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 the 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-

	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.	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 verification and validation and uncertainty analysis is available for review. Additionally, the NRC 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	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 TR, which will be submitted prior to the OL application. Prior to any fabrication effort, a Manufacturing Quality Plan will be created to identify the	With respect to the limitation's point on fuel specifications, the NRC staff reviewed the preliminary fuel specifications for both the LUPs and HUPs in the 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

these are not incorporated within the scope of the TR.	methods to verify all specified properties of produced fuel product and the associated sampling statistics to apply quality controls to a produced fuel batch. PSAR section 6.4.2.3 discusses the Quality Control/QA procedures for manufacturing.	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 the construction, which is sufficient to address this portion of the limitation at the CP stage.
Electric Power Research Institute (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, the 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. The 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 its 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), silicone carbide (SiC) thickness, SiC density, and outer pyrolytic carbon (OPyC) density. AGR-1	The NRC staff finds that 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.

	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 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,	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 for additional discussion on these planned tests.	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.

considering uncertainty in both the AGR test results as described in the TR and any analytical uncertainty resulting from the proposed analytical method.	These test results will be summarized in a future revision of the TRISO-X Pebble Fuel Qualification Methodology TR provided as part of 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, the NRC staff reviewed the list of radionuclides LME identified as important to dose. The NRC staff finds this list to be reasonable based on a comparison to historical HTGRs and a confirmatory analysis performed using SCALE. Based on this confirmation, the NRC 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 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.

The system-specific regulatory requirements applicable to the CMS are as follows:

- 10 CFR 50.34(a)(5).

The system-specific regulatory guidance applicable to the CMS is as follows:

- RG 1.87, Revision 3, “Acceptability of ASME Code Section III, Division 5, ‘High Temperature Reactors,’” (ML25176A084).

- RG 1.26, Revision 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 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 that the CMS supports performance of RSF 1.1.2.1 to provide means to insert negative reactivity to ensure safe shutdown to maintain long-term subcriticality, RSF 1.2.1.1 to transfer heat from fuel to pressure vessel to maintain passive core heat removal, and RSF 1.4.1.2 to 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, and the results are summarized in PSAR 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 to control core reactivity to preclude reactor trips by controlling reactivity with control rods and 1.2.2.2 to 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, Revision 3. The NRC staff finds 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 **[**

]].

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 PSAR 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 QA requirements under PDC 1 for the SR aspects of the CMS. 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 above NRC staff's evaluation, the NRC staff finds that the preliminary design of the CMS is consistent with 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 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 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 PSAR 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 is consistent with 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 graphite core assembly (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 is consistent with 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 13.1 of this SE. The NRC staff evaluated the environmental compatibility assessment method in section 3.8 of this SE and found that it is consistent with PDC 4.

The NRC staff also reviewed PSAR section 6.4.3 to assess whether it is consistent with PDC 4 requirements for 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 with 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 is consistent with PDC 4. The NRC 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 finds it acceptable (see section 3.7 of this SE). Based on the discussion above, the NRC finds that the preliminary design of the CMS system is consistent with 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 reliability and integrity management (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 is consistent with PDC 6, with the 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 TR, previously approved by the NRC staff. As discussed in the basis for PDC 10 in the Xe-100 PDC TR, 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 Systems (PDC-RFDC 26 and PDC-CDC 26)

The NRC staff reviewed PSAR section 6.4.10, which states that LMGS 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 graphite core components (GCC). 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, is consistent with 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 is consistent with 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, is consistent with 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 be consistent with PDC-RFDC 70, to maintain the core geometry for sufficient heat removal and rod insertion. The NRC 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 that the CMS supports the performance of RSF 1.1.2.1 to provide means to insert negative reactivity to ensure safe shutdown to maintain long-term subcriticality, RSF 1.2.1.1 to transfer heat from fuel to pressure vessel to maintain passive core heat removal, and RSF 1.4.1.2 to 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, PDC-RFDC 34, and PDC-RFDC 70 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 PSF 1.1.3.1 to control core reactivity to preclude reactor trips by controlling reactivity with control rods and NSRST PSF 1.2.2.2 to maintain helium flow path to maintain core heat removal through active means. LME identified PDC-CDC 26 and PDC-CDC 34 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.

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 TS 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 NRC staff's findings are based on the variables, conditions, or items that are likely to become subjects of TS 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 TS 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 applicant meeting the following:

- 10 CFR 50.34(a)(3)(i) requirements for the CMS by identifying PDCs applicable to the CMS and describing the preliminary design of the CMS in PSAR section 6.4.3.1;

- 10 CFR 50.34(a)(3)(ii) requirements by describing how the preliminary CMS design meets the requirements of the identified PDCs in PSAR sections 6.4.3.1 through 6.4.3.3;

- 10 CFR 50.34(a)(3)(iii) requirements 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 in PSAR sections 6.4.3.1 through 6.4.3.3; and

- 10 CFR 50.34(a)(5) requirements by identifying probable subjects of TS for the CMS in part IV of the CP application.

6.4.4 Graphite Core Assembly

PSAR section 6.4.4 describes the GCA, which is constructed of individual 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.

The system-specific guidance for the evaluation of the GCA is as follows:

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

- 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 TR, "TR009380, Revision A, Xe-100 Graphite Core Assembly Material Qualification and Design Methodologies," ML26114A162 (hereafter referred to as the GQDM TR). 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. The L&Cs in the SE for the GQDM TR are addressed by LME in PSAR table 1.4-14 and evaluated in 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 PSAR section 6.4.4.1.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

The GCA is classified as SR. The GCA partially conforms to the 2023 Edition of ASME BPVC section III, division 5, as endorsed with conditions in RG 1.87, Revision 3. The GQDM TR describes the areas of partial conformance, and the NRC staff's evaluation of the GQDM TR is documented in the SE for the GQDM TR. 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 NRC staff's 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 QA 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 NRC staff's endorsement of ASME BPVC III.5 in RG 1.87, Revision 3, the NRC staff finds that the preliminary design of the GCA is consistent with 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 3.7.1 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. Condition 17 in the SE for the GQDM TR addresses the material properties of the irradiated graphite necessary for determining the seismic response of graphite to meet PDC 2. Based on the discussion above, the NRC staff finds that the information provided in the PSAR is consistent with PDC 2 at the CP stage.

Fire Protection (PDC 3)

PSAR section 6.4.1.1 references the GQDM, which states that the graphite used to fabricate the individual GCCs to construct the GCA meets 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 HPB, 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 NRC staff finds that the GCA is consistent with 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. 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 finds the methods acceptable for all conditions in the NRC staff's evaluation 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 the SE.

The radiation effects on graphite material properties were accounted for in the Flownex and GOTHIC models, as described in PSAR section 3.1.3. As stated in PSAR section 3.1.3, 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 the 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 of this SE for the MELCOR confirmatory calculations. Therefore, the NRC staff finds that the GCA is consistent with PDC 4.

Sharing of Structures, Systems, and Components (PDC 5)

GCA SSCs are not shared between reactor modules. Therefore, the NRC staff finds that the GCA is consistent with 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 that PSAR section 8.9 states 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's review finds 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 a 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 staff finds that the preliminary design of the GCA is consistent with 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 NRC staff's evaluation of the GQDM TR, as conditioned, the NRC staff finds that the GCA design is consistent with 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 that the GCA is designed to provide channels for the insertion path of the RCSS rods. The 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 finds the methodology to be an acceptable approach to ensure that the margin to failure is maintained through design life.

The NRC staff determined that the preliminary GCA design is consistent with PDC-CDC 26 and PDC-RFDC 26 because the insertion path for the RCSS 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 NRC staff finds that the GCA is consistent with PDC-CDC 34.

Residual Heat Removal (PDC-RFDC 34)

PSAR section 6.4.4 states that 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 is consistent with 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 finds that the preliminary GCA design is consistent with 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 RSF 1.1.2.1 to provide means to insert negative reactivity to ensure safe shutdown to maintain long-term subcriticality, RSF 1.2.1.1 to transfer heat from fuel to pressure vessel to maintain passive core heat removal, and RSF 1.4.1.3 to maintain acceptable GCA geometry. LME identified PDC-RFDC 26, PDC-

RFDC 34, and PDC-RFDC 70 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 and PDC-CDC 34 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 TS 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 NRC staff finds that the preliminary information regarding TS for the GCA addresses the requirements of 10 CFR 50.36(c)(4), "Design features."

In addition, the NRC staff finds that the information provided meets the requirements of 10 CFR 50.34(a)(5). The NRC staff's findings are based on the variables, conditions, or items that are likely to become subjects of TS 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 TS 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 applicant meeting the following:

- 10 CFR 50.34(a)(3)(i) requirements for the GCA by identifying PDCs applicable to the GCA in PSAR section 6.4.4.1;

- 10 CFR 50.34(a)(3)(ii) requirements by describing how the preliminary GCA design meets the PDCs identified for the GCA in PSAR section 6.4.4.1;

- 10 CFR 50.34(a)(3)(iii) requirements 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 in PSAR section 6.4.4.1; and

- 10 CFR 50.34(a)(5) requirements by identifying probable subjects of TS for the CMS in part IV of the CP application.

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 FHS via a portion of the defuel chute pressure vessel (DCPV). The DCPV surrounds the FUCH and provides the 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.

The 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).

The system-specific guidance for the evaluation of the FUCH is as follows:

- DANU-ISG-2023-01 “Material Compatibility for non-Light Water Reactors,” (ML22203A175).
- RG 1.87, Revision 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-RFDC 34 and PDC-CDC 34 – Residual heat removal
- PDC-RFDC 70 – Reactor 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 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, R&D 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 RCCS. The FUCH includes a design feature to support active cooling with cooled primary helium flow from the SGs.

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, Revision 3, are designed to ASME BPVC section III (2023), division 5, subsection HG, subpart B. Subsection HG, subpart B, is the appropriate subsection for the 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 and table 13.3-1. The scope of the R&D program for addressing areas requiring code supplementation for the FUCH includes the extension of qualified life. The preliminary FUCH design temperature is within the allowed limits of ASME.III.5.HGB, but the 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

QA 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's evaluation, including the application of a 10 CFR Part 50, Appendix B QAP, the endorsement of ASME.III.5 in RG 1.87, Revision 3, and the inclusion of the FUCH in the R&D program, the NRC staff finds that the preliminary design of the FUCH is consistent with 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 that 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 SSE without losing the capability to perform its RSFs. 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 NRC 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 10 CFR Part 50, Appendix S, it is designed to withstand the effects of the SSE and remain functional per PSAR section 3.7. The NRC staff finds that 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 HPB and is constructed of metallic materials that are inherently noncombustible and non-fire sustaining. Based on the information provided in the PSAR section 6.4.5, the NRC staff finds that the preliminary design is consistent with PDC 3 because the FUCH is a passive component made of noncombustible materials and is contained within the HPB.

Environmental and Dynamic Effects Design Bases (PDC 4)

The NRC 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 13.1 of this SE. The NRC staff also reviewed PSAR section 6.4.5 to assess whether the approach to protecting the FUCH against dynamic effects is consistent with PDC 4. 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 NRC staff will review the FUCH design described in the FSAR to confirm compliance with PDC 4 regarding sufficient protection of the 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 NRC staff reviewed arrangement drawings and confirmed that the functions of the FUCH are not shared between modules. Therefore, the NRC staff finds that the FUCH is consistent with 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 NRC 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 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 is consistent with 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 NRC 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 NRC 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 NRC staff assesses in its evaluation of heat removal from the fuel within the FUCH. Based on the above assessment of the FUCH design, the NRC staff finds that the PSAR provides sufficient detail to describe how the FUCH is consistent with 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 the 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 that 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 RCCS. The NRC staff finds that 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 NRC staff evaluated the design bases of the FUCH against the criteria of PDC-RFDC 34 and PDC-CDC 34. The NRC staff finds that the FUCH 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 the audit and compared it with the preliminary configuration of the 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 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 NRC staff finds that the preliminary design bases for passive heat removal is consistent with 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 NRC staff finds is consistent with PDC-CDC 34.

Reactor Structural Design Basis (PDC-RFDC 70)

The NRC staff evaluated the design basis against PDCs 1, 2, and 6 and determined that the preliminary design of the FUCH has adequate structural integrity because the design and construction code in ASME section III, division 5, subsection HG, as endorsed by RG 1.87, Revision 3; 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 the relationship to PDC 4 above, the environmental compatibility of the materials has been identified for further R&D and design development to support the 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 the 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 the design development to support the 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 the performance of RSF 1.2.1 and RSF 1.4.1. 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 6.4.5.1 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 NRC staff finds that the preliminary design supports the RSFs with appropriate criteria for reliability and capability defined. The NRC 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) R&D 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 the performance of PSF 1.2.2 to maintain core heat removal through active means and subsidiary PSF 1.2.2.2 to restore active heat removal. The FUCH includes a design feature to support active cooling with cooled primary helium flow from the SGs. This feature accommodates the establishment and restoration of primary cooling flow to the FUCH following AOOs and provides assurance that SARRDLs are not exceeded.

The NRC staff finds that 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 RG 1.253, because the functions have been defined with preliminary performance requirements supported by plans for R&D 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 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 that the information provided meets the requirements of 10 CFR 50.34(a)(5). The NRC staff’s finding is based on the variables, conditions, or items that are likely to become subjects of TS for the FUCH being identified. Consistent with 10 CFR 50.34(b)(6)(vi), the NRC staff will evaluate the proposed TS when detailed design information is available at the OL application stage.

Research and Development

Paragraph 50.34(a)(8) of 10 CFR requires the identification of any facility SSCs that necessitate R&D to confirm the adequacy of their design. PSAR table 13.3-1 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 NRC 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 the 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 R&D affecting the FUCH;

- meets the requirements of 10 CFR 50.35(a) regarding the incomplete qualification of the FUCH for environmental conditions under PDC 4 and PDC-RFDC 70, that is reasonably expected to be resolved through R&D and addressed in the OL application;

- meets the requirements of 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 the requirements of 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 the requirements of 10 CFR 50.34(a)(5) by identifying key design features of the FUCH as probable subjects of TS;

- meets the requirements of 10 CFR 50.34(a)(8) by identifying the R&D needs for the FUCH to confirm the performance of their metallic materials in the Xe-100 environmental operating conditions; and

- meets relevant portions of 10 CFR 50.34(a)(12) by identifying that the FUCH is designed to retain its function for the SSE.

6.4.6 Reactor Pressure Vessel

PSAR section 6.4.6 describes the 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 (CV) 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 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.

The system-specific regulatory requirements applicable to the RPV are as follows:

- 10 CFR 50.34(a)(5).

The system-specific guidance applicable for the evaluation of the RPV is as follows:

- RG 1.87, Revision 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 PSAR 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, Revision 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 is consistent with PDC 1 by using and supplementing appropriate codes and standards commensurate with the safety significance of the component safety functions. In addition, the NRC staff finds that 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 QA 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 is consistent with 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 that 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 that 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 NRC staff considered the DBHL information described in PSAR chapter 2 and PSAR section 6.1. The NRC staff's 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 RB would not pose a flooding hazard for the RPV.

Based on the evaluation above, the NRC staff finds that the RPV design is consistent with 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 FPP, 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's evaluation of the LME FPP 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 is consistent with 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 finds that it is consistent with 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 is consistent with PDC 4. As discussed in section 6.1 of this SE, the NRC staff finds that LME's approach is appropriate to ensure that the SR SSCs will be adequately protected against the dynamic effect, including HELB 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 is consistent with 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 NRC 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 is consistent with 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 NRC 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 finds 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 is consistent with 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 TR. As discussed in the basis for PDC 10 in the Xe-100 PDC TR, 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 that the 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 instrumentation and control (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, and is consistent with 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 is consistent with 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 the 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 are consistent with 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, consistent with 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 RCSS consists of two separate rod banks, the RCS control rods and the 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 18 CRDM housings that interface with the 18 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 is consistent with 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 is consistent with 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 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 by PDC 31. The NRC staff finds that 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 is consistent with 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 NRC 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 of 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 OL application stage.

Based on the above information, the NRC staff finds that the preliminary design of the RPV is consistent with 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 **[[ ]]** 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 NRC 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 is consistent with PDC-RFDC 70.

Safety Functions

PSAR section 6.4.6 states that the RPV is designed to support RSF 1.1.2.1 to provide means to insert negative reactivity to ensure safe shutdown, RSF 1.2.1.2 to transfer heat from vessel to reactor cavity, and RSF 1.4.1.2 to 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 to control core reactivity to preclude reactor trips and PSF 1.2.2.2 to restore active heat removal, as well as PSFs 2.7.1, 2.7.2, and 2.7.3 to 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 TS 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 a 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 that the information provided meets the requirements of 10 CFR 50.34(a)(5). The NRC staff's finding is based on the variables, conditions, or items that are likely to become subjects of TS 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 TS when detailed design information is available at the OL application stage.

6.4.6.2 Conclusion

The NRC 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 applicant meeting the following:

- 10 CFR 50.34(a)(3)(i) requirements for the RPV by identifying PDCs applicable to the preliminary design of the RPV and describing the preliminary design of the RPV in PSAR section 6.4.6.1;

- 10 CFR 50.34(a)(3)(ii) requirements by describing how the preliminary design of RPV meets the PDCs identified in PSAR sections 6.4.6.1 and 6.4.6.2;

- 10 CFR 50.34(a)(3)(iii) requirements 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 in PSAR sections 6.4.6.1 and 6.4.6.2; and

- 10 CFR 50.34(a)(5) requirements by identifying probable subjects of TS for the RPV in part IV of the CP application.

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.

The system-specific guidance applicable for the evaluation of the RVSS is as follows:

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

- RG 1.26, Revision 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, 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, Revision 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 QA 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 is consistent with 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 NRC staff considered the DBHL information described in PSAR chapter 2 and section 6.1. The NRC 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 is consistent with 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 FPP, 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 is consistent with 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 finds that it is consistent with 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 is consistent with PDC 4. Per section 6.1 of this SE, the NRC staff finds that LME's approach is appropriate to ensure that the SR SSCs will be adequately protected against the dynamic effect, including HELB 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 is consistent with PDC 4. 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.

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 finds it acceptable as discussed in section 3.7 of this SE. Based on the discussion above, the NRC finds that the preliminary design of the RVSS system is consistent with 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 is consistent with PDC 6, with review of the RIM program reasonably being left to the OL application stage.

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 that 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 is consistent with PDC-RFDCs 70 and 71.

Safety Functions

PSAR section 6.4.7 states that the RVSS is designed to support RSF 1.4.2. LMGS identified PDC-RFDC 71 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 NRC 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 applicant meeting the following:

- 10 CFR 50.34(a)(3)(i) requirements for the RVSS by identifying PDCs applicable to the preliminary design of the RVSS and describing the preliminary design of the RVSS in PSAR section 6.4.7.1;

- 10 CFR 50.34(a)(3)(ii) requirements by describing how the preliminary design of RVSS meets the PDCs applicable to the preliminary design of the RVSS in PSAR section 6.4.7.1; and

- 10 CFR 50.34(a)(3)(iii) requirements 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 in PSAR section 6.4.7.1.

6.4.8 Defuel Chute Pressure Vessel

PSAR section 6.4.8 describes the 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 Y-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 SGs 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 SUMs, 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).

The 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 TS.

- 10 CFR 50.34(a)(12), which requires compliance with the earthquake engineering criteria in 10 CFR Part 50, Appendix S.

The system-specific guidance for the evaluation of the DCPV is as follows:

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

- 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 HPB from the bottom head of the RPV to the 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 Y-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 SUMs (one at each end) by flanged connections. The Y-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 PSF 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 RB 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 SGs 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, Revision 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 is consistent with 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 QA 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 is consistent with 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 that 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 SSE without losing the capability to perform its RSFs. 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 NRC 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 10 CFR Part 50, Appendix S, 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 preventive measure. Additionally, the NRC staff review determined that the DCPV is at an elevation that is significantly higher than the site's still water flood elevation. Therefore, the NRC 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 NRC 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 noncombustible 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 NRC staff finds that the preliminary design is consistent with PDC 3 because the DCPV is a passive component made of noncombustible 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 NRC staff also reviewed PSAR section 6.4.8.1 to assess whether the facility design is consistent with 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 NRC 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 is consistent with 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 NRC staff finds the description sufficient to determine the DCPV is not a shared SSC between multiple nuclear power units and therefore is consistent with 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 NRC 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 is consistent with 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 stage.

Reactor Design (PDC 10)

The NRC 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 that the SARRDLs are not exceeded; and (2) providing adequate heat removal for the fuel within the FUCH to ensure that 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 the 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 NRC 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 NRC staff finds that the PSAR provides sufficient detail to describe how the DCPV is consistent with 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 NRC staff evaluated the information provided in PSAR section 6.4.8.1 related to I&C. The NRC staff recognizes that continued circulator operation without active heat removal in the SGs, 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 investment protection system (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 NRC staff finds that the applicant has described suitable I&Cs 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 is consistent with 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 is consistent with PDC14, PDC-CDC 15, PDC-CDC16, and PDC 31.

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

The NRC staff reviewed PSAR section 6.4.8.1, which states that the features that support the 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 NRC 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 the 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 that the analysis demonstrated that the FUCH and DCPV peak temperatures during events involving a loss of forced cooling would remain below the 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 RCCS. The DCPV supports active cooling of the fuel by providing a flow path for cooled helium from the SG 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 NRC 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 the 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 the changes in the 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 is consistent with 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 NRC staff finds is consistent with 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 that the preliminary DCPV is consistent with PDC-RFDC 70 because the design and construction code is ASME section III, division 5, subsection HBA, as endorsed by RG 1.87, Revision 3; the DCPV design is subject to the criteria for the 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 and RSF 1.4.1. 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 1.2.1.2 – 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 that the 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 NRC staff finds that the preliminary design supports the RSFs with appropriate criteria for reliability and capability defined. The NRC staff also finds that the applicant describes the 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 to maintain core heat removal through active means and subsidiary PSF 1.2.2.2 to restore active heat removal. The DCPV design includes a feature to support active cooling with cooled primary helium flow from the SGs. As evaluated above regarding DCPV design basis conformance with PDC-CDC 34, the NRC staff finds that this feature accommodates the 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 the performance of PSF 2 to retain radionuclides in the HPB and PSF 2.7 to 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 NRC 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 TS for the DCPV in part IV of the CP application. The proposed LCOs identified in table IV-3 of the proposed TS relate to RB 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 NRC staff reviewed information provided in the PSAR related to the proposed TS parameters and conditions. The proposed parameters of RB 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 LBEs 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 that 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 NRC 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 the requirements of 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 the requirements of 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 the requirements of 10 CFR 50.34(a)(5) by identifying preliminary FHS variables and conditions that are probable subjects of TS; and

- meets relevant portions of 10 CFR 50.34(a)(12) by identifying that the DCPV is designed to retain its function for the SSE.

6.4.9 Primary Pressure Relief System

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, and rupture discs connected by piping. The rupture disc connections are above the elevation of the expected water level following a SG tube rupture (SGTR). The PPRS connects to the SG 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 over-pressurization to ensure that sufficient pressure relief capacity is provided to maintain core geometry within acceptable limits.

The system-specific regulatory requirements applicable to the PPRS are as follows:

- 10 CFR 50.34(a)(5)

The system-specific guidance applicable for the evaluation of the PPRS is as follows

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

- RG 1.26, Revision 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 is classified as SR based on its function to provide sufficient pressure relief capacity to maintain core geometry within acceptable limits during DBEs and DBAs. The PPRS supports performance of RSF 1.4.1.1 to control primary pressure. The safety functions that the PPRS 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.

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, Revision 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 percent volumetric examination (UT or RT) of pressure boundary fusion welds and requiring full penetration boundary fusion welds. The NRC staff finds that 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 the requirements of 10 CFR Part 50, Appendix B, and is appropriate for meeting the QA 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 QAP described in part III of the LMGS QAPD and PSAR section 8.1. Additional special treatments related to QA will be specified at the OL application stage. The application of part III of the LMGS QAPD at the CP stage is sufficient as a preliminary special treatment to meet the QA 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 is consistent with 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. The NSRST components of the PPRS are categorized as seismic category IIA to ensure that 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's 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 is consistent with 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 SG 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 FPP, 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 is consistent with 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 finds 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 finds that it is consistent with 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 is consistent with PDC 4. The NRC 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 is consistent with 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 SR 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 a 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 NRC 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 inservice testing (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.

As confirmed by the 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 the NRC 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 reliability, availability, and maintainability (RAM) program, which is discussed in SE section 8.10. 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 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, 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 are consistent with 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 (DID) adequacy.

Reactor Design (PDC 10)

Per the NRC-approved Xe-100 PDC TR, PDC-10 is a higher-level requirement that is met via application of PDC-CDC 26 and PDC-CDC 34. The PPRS protects the RPV from damage in a pressure transient, which could adversely impact the geometry of the core and the functions associated with it, as discussed below in PDC-RFDC 70. This includes aspects related to the ability to insert negative reactivity using the control rods and the ability to remove heat through active and passive means. So, while PDC-CDC 26 and PDC-CDC 34 are not applied to the PPRS, the functions performed support other SSCs performing functions associated with these PDCs. 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 that the preliminary design of the PPRS meets the requirements of PDC 10 by controlling primary pressure to provide overpressure protection for the RPV system, which supports insertion of negative reactivity and active heat transfer. Therefore, the NRC staff finds that the preliminary design of PPRS is consistent with 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 is consistent with 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 is consistent with 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 the structural integrity of the PPRS. In addition, the NRC staff noted that the rupture discs and pressure relief valves directly vent to the SST 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, consistent with 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 is consistent with 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 NRC 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 is consistent with 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, and is consistent with PDC-RFDC 70.

Safety Function

PSAR section 6.4.9 states that the PPRS is designed to support RSF 1.4.1. LMGS identified PDC-RFDC 70as 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 PSF 2.7.1 to control the frequency of small depressurization events, PSF 2.7.2 to control the frequency of medium depressurization events and PSF 2.7.3 to 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 are consistent with the PDCs as evaluated above in this SE section.

Technical Specifications

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

Based on the review of the information provided in the CP application, as discussed above, the NRC staff finds that the information meets the requirements of 10 CFR 50.34(a)(5). The NRC staff's finding is based on the variables, conditions, or items that are likely to become subjects of TS 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 TS 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 applicant meeting the following:

- 10 CFR 50.34(a)(3)(i) requirements for the PPRS by identifying PDCs applicable to the PPRS and describing the preliminary design of the PPRS in PSAR section 6.4.9.1;

- 10 CFR 50.34(a)(3)(ii) requirements by describing how the PPRS meets the PDCs identified in PSAR sections 6.4.9.1 and 6.4.9.2;

- 10 CFR 50.34(a)(3)(iii) requirements 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 in PSAR sections 6.4.9.1 and 6.4.9.2; and

- 10 CFR 50.34(a)(5) requirements by identifying probable subjects of TS for the PPRS in part IV of the CP application.

6.4.10 Reactivity Control and Shutdown System

The NRC staff reviewed PSAR section 6.4.10, which describes the 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 subcritical 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.

The 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:

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

- RG 1.87, Revision 3, “Acceptability of ASME Code Section III, Division 5, ‘High Temperature Reactors,’” (ML25176A084).

- 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 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 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” 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, 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 the 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 3.8.1 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 the requirements of 10 CFR Part 50, Appendix B, and is appropriate for meeting the QA 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 QAP described in part III of the LMGS QAPD and in PSAR section 8.1. Additional special treatments related to QA 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 QA 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 RXST and NSRST 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 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 noncombustible 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 noncombustible 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 noncombustible 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 NRC staff finds that 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 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 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 NRC 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 finds 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 SR 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 the material qualification for NAR from internal gas release of the B_4C during the service lifetime. The PSAR further identifies the FHS, CV and 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 to be 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 is consistent with PDC 4. The NRC 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 finds 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 preservice and inservice 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 PSAR section 8.10. The RCSS design provides access for implementing remote preservice and inservice 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 NRC staff finds that 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 preoperational testing and startup testing described in PSAR sections 12.1.1.2 and 12.1.1.3, respectively. The evaluation of the preoperational 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 DID adequacy. Based on the discussion above, the NRC finds that the preliminary design of the RCSS is consistent with 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 the 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, Revision 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 the 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 the performance of RSF 1.1.2 to maintain long-term subcriticality with no reliance on operator action or electrical power. PSAR table 3.1-2 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 is consistent with PDC-RFDC 26.

Reactivity Control Systems (PDC-CDC 26)

PDC-CDC 26 is aligned with the performance of NSRST PSF 1.1.3. The RCSS is designed to insert NARs 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 3.1-2 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 distributed control system (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 is consistent with 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 I&Cs 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 is consistent with 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 the use of the appropriate ASME codes and standards as the primary means of assuring HPB integrity and support performance of NSRST PSF 2.7 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 is consistent with 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 to maintain long-term subcriticality, RSF 1.1.2.1 to maintain long-term subcriticality during safe shutdown conditions, and NSRST PSF 1.1.3 to 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 is consistent with 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 the application of PDC-CDC 13 as a design requirement applied to the interfacing systems such as nuclear instrumentation system (NIS) and IPS. NIS is an SR system described in PSAR section 6.4.14 and IPS is an 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 is consistent with 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 are 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 the 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 is consistent with 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 to support the performance of RSFs 1.1.2 and 1.1.2.1.

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

- The RCSS neutron absorbers are designed with 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 the 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 (ML26293A340), the NRC staff inquired about RSF 1.4.1 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 finds 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 finds, 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 the 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 Testing

Criterion XI, “Test Control,” of 10 CFR Part 50, Appendix B 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 components as incorporating FOAK technology, design or configuration and thus did not identify any FOAK special treatment tests to demonstrate the capability of the RCSS to perform necessary RSF and NSRST PSFs or to meet 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 the 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.

Through the audit, LME indicated that:

- The segmented NAR is not unique to the Xe-100 design. Segmented rod designs have been used in other high-temperature gas reactors (HTGRs) 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 a chain sprocket.

- The stacked boron carbide ring has been used in other HTGRs including the HTTR and HTR-10, with both designs reporting five compacts being used inside each segment. The design of the 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 AGRs in the United Kingdom (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 HTGRs, and the helium coolant with graphite moderated environment is well established in HTGR 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 finds 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 TS 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 TS includes a minimum number of rods per bank to support RSF 1.1.2 to be fulfilled during specific LBEs. At the CP stage, the PSAR identified no 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 that the information provided meets the requirements of 10 CFR 50.34(a)(5). The NRC staff's finding is based on variables, conditions, or items that are likely to become subjects of TS for the RCSS being identified. Consistent with 10 CFR 50.34(b)(6)(vi), the NRC staff will evaluate the proposed TS when detailed design information for the RCSS is available at the OL application stage.

Research and Development

Paragraph 50.34(a)(8) of 10 CFR requires the identification of any facility SSCs that necessitate R&D to confirm the adequacy of their design. PSAR table 13.3-1 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 is met. The audit response further pointed to the 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 preoperational and start-up test programs, which include RCSS testing.

Based on the review of the RCSS design presented in PSAR section 6.4.10 and confirmatory discussions from the 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 the 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 the requirements of 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 the requirements of 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 the requirements of 10 CFR 50.34(a)(5) by identifying key RCSS variables and conditions as probable subjects of TS;

- meets the requirements of 10 CFR 50.34(a)(8) by identifying the R&D needs for RCSS to confirm the performance of their metallic materials in the Xe-100 environmental operating conditions; and

- meets the requirements of 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.

6.4.11 Reactivity Cavity Cooling System

PSAR section 6.4.11 describes the 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 (356 mm) outer diameter) connecting the toroidal downcomer header to the riser header. Two boil-off tanks are connected above each downcomer header. Both the boil-off 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. DID makeup options (nuclear island fire protection, nuclear island process water system, condensate system, 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 percent 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. 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.

The 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 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 that the SFA portion of the RCCS is appropriately categorized as an 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 an 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 QA 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 QAP described in part III of the LMGS QAPD and in PSAR section 8.1. Additional special treatments related to QA 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 QA

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 RXST and NSRST 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 DBHLs 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 NRC 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 QAP 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 finds 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 the 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, administrative controls, personnel training, and maintenance programs. PSAR

section 8.6 states that the operational FPP will conform to RG 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); and
- 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 NRC 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 NRC 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 the development of temperature fields for bounding transient and the 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 NRC 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 is consistent with 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 the 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 found 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 RSFs during AOOs, DBEs, and DBAs in any individual module. Based on the information above, the NRC staff finds that the RCCS design is consistent with 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. The 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 the inspection addresses credible degradation mechanisms, the NRC 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 degradation mechanism assessment 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 that the 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 the 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 Post Event Monitoring System (PEMS) for online monitoring. The adequacy of the RCCS I&C design is addressed under the evaluation for PDC-CDC 13.

Based on the discussion above, the NRC staff finds that the SR SSCs are consistent with 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 DID 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 the application of PDC-CDC 34, which is aligned with the 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 that the discussion is consistent with the NRC staff-approved Xe-100 PDC LTR, Revision 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 that 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.

Instrumentation and Control (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 MCR through the 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 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 NRC staff finds that the proposed instrumentation is adequate to monitor the performance of the NSRST RCCS and is consistent with 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 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 pressurized loss of forced cooling (PLOFC) and depressurized loss of forced cooling (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 exclusion area boundary 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 are 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 (EMs) 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 RSFs, 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 (ML26293A340), the NRC 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 the 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 the 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 will provide the NRC staff with reasonable assurance that the required information will be supplied in the FSAR and therefore support the 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 the 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 finds 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 EM, 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.

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.

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 sections 3.3 through 3.5 demonstrate the 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 the 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 NRC staff finds that the preliminary NSRST RCCS design information is consistent with PDC-CDC 34.

Preliminary Analysis of RCCS

Paragraph 50.34(a)(4) of 10 CFR 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 NRC staff notes that the PSAR does not clearly demonstrate the fidelity of the RCCS EM 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 EM development and assessment process 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 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 NRC 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 NRC 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 EM 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))

Paragraph 50.34(a)(5) of 10 CFR requires identification and justification of variables, conditions, or items that, based on preliminary safety analysis, are likely to become subjects of TS for the facility. CPA part IV, table IV-3 identifies preliminary systems subject to LCOs, including the requirement to maintain sufficient water inventory in the RCCS boil-off tanks to ensure that the SR portion of the RCCS can perform its RSFs. The NRC staff finds that the 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 NRC 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 TS 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 EM 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 the issuance of a CP in accordance with 10 CFR 50.35(a).

FOAK Testing

Criterion XI, "Test Control," of 10 CFR Part 50, Appendix B 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 EM. 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 finds 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 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 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 NRC 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 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 TS.

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 EM 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 conforms to the design basis during the evaluation of the FSAR as part of the OL application review.

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 start-up and shutdown system during certain LBEs. The SGS consists of the SG, steam generator feedwater system (SGFS), and the steam generator steam system (SGSS).

The system-specific regulatory guidance applicable to the SGS is as follows:

- RG 1.87, Revision 3, “Acceptability of ASME Code Section III, Division 5, ‘High Temperature Reactors,’” (ML25176A084).
- RG 1.26, Revision 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 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. 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 helium services system (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, Revision 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 the 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 a 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 is consistent with 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, Revision 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 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 NRC 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 is consistent with 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 the requirements of 10 CFR Part 50, Appendix B, and is appropriate for meeting the QA 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 QAP described in part III of the LMGS QAPD and in PSAR section 8.1. Additional special treatments related to QA will be specified at the OL application stage. The application of part III of the LMGS QAPD at the CP stage is sufficient as a preliminary special treatment to meet the QA 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 RB 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 nuclear island auxiliary structure (NIAS) and the SST, or between the RB and NIAS, and these SSCs will be designed to resist DBHLs 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 NRC 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 that 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 stage.

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 noncombustible metallic materials. Likewise, the insulation used in the SGS is made of high-temperature-resistant and inherently noncombustible ceramic material. The SGS is located in the SG 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 NRC staff in section 6.4.1 of this SE. In addition, PSAR section 6.4.12 states that the SGS is protected by fire protection systems and a comprehensive plant 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 is consistent with 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 inservice 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 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 the evaluation of external pressure loads from rapid compartment pressurization, and the 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 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's 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 is consistent with PDC 4. The NRC 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 is consistent with 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 a 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 NRC 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.

As confirmed by the 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 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 NRC 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 to be 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, will be performed to provide a baseline for the inservice inspections, which is evaluated in SE section 12.1.

Based on the discussion above, the NRC staff finds that the preliminary design of the SR SGS SSCs is consistent with 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 DID 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 NRC staff finds that the SGS is consistent with 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’s 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 helium purification system (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.

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)

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 finds 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 NRC staff's 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 the 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 is consistent with 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 SRVs 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 is consistent with 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 finds the application of these PDCs to the selected SGS components to be 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 I&C 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 is consistent with PDC-CDC 30.

Fracture Prevention of Reactor Helium Pressure Boundary (PDC 31)

PDC 31 requires that the reactor HPB 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, Revision 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 the 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 that the SGS is consistent with PDC 31 based on the evaluation above and the results of the NRC staff's 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 that the preliminary design of the SGS is consistent with PDC-CDC 34.

Safety Functions

PSAR section 6.4.12.1 states that the SGFS's NIFI and SGSS's NISI valves support the performance of RSF 1.3.1. In PSAR table 5.3-1, LME identified PDC-RFDCs 13 and PDC-RFDC 30 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 SGS shall be designed to isolate the main feedwater system from the SG 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 SGS shall be designed to isolate the main steam system from the SG 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, PSF 1.3.2, PSF 1.3.3, PSF 2.2.1, and PSFs 2.7.1-2.7.3. 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 AOOs, DBEs, or DBAs.

- 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 NRC 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 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 TS for the SGS based on the preliminary design and safety analysis. The proposed list of LCOs identified in the proposed TS 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 that the preliminary information regarding TS for the SGS addresses the requirements of 10 CFR 50.36(c)(2), "Limiting Conditions for Operations." At the CP stage there are no SLs 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 that the information provided for the SGS meets the requirements of 10 CFR 50.34(a)(5) because the variables, conditions, and items that are likely to become subjects of TS for the SGS are identified. Consistent with 10 CFR 50.34(b)(6)(vi), the NRC staff will evaluate the proposed TS 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 TS.

6.4.13 Reactor Protection System

PSAR section 6.4.13 describes the 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 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 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 TR, TR-1015-18653, “Design of the Highly Integrated Protection System Platform,” Revision 2 (it ML17256A892-Public, ML17256A894-Proprietary) has been IBR 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 Institute of Electrical and Electronics Engineers (IEEE) 603-1991 and the correction sheet dated January 3, 1995.

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/International Society of Automation (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

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 QA requirements to comply with PDC 1 and LME's QAP, as described in PSAR section 8.1. Additionally, the NRC staff reviewed PSAR 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 PSAR table 6.4.13-1 as applicable to the RPS and supports meeting 10 CFR 50.55a(h)(3). The NRC staff's evaluation of IEEE 603, as it relates to the RPS, is discussed later in this section of the SE. Based on the information provided in the PSAR, the NRC staff finds that the preliminary design of the RPS is consistent with PDC 1 by using nuclear standards endorsed in NRC regulatory guidance and the LME QAP.

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 is consistent with 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 NRC staff finds that the preliminary design of the RPS is consistent with 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 NRC staff finds that locating the RPS in the FHAB is consistent with 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 is consistent with 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)(3); and IEEE 7-4.3.2-2016, as endorsed by RG 1.152. The RPS also has self-test features and is subject to TS and periodic surveillances. Finally, preoperational testing and startup testing is described in PSAR section 12.1.1 and evaluated in section 12.1 of the SE. Based on the NRC staff's review of the information provided in the PSAR, the evaluations in section 8.3 and 12.1 of this SE, and by using nuclear standards endorsed in NRC regulatory guidance, the NRC staff finds that the preliminary design of the RPS is consistent with 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 RCS 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 that the preliminary design of the RPS is consistent with 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 I&Cs 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 NRC staff finds that the preliminary design, to include specified manual switches to initiate prompt safe shutdown outside the MCR, is consistent with 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. The PSAR states that design of the RPS is in accordance with IEEE 603-2018, which supports meeting the requirements of 10 CFR 50.55a(h)(3). Ensuring that design limits are not exceeded and RSFs are achieved is accomplished by having I&C, as described above in the evaluation of PDC-RFDC 13. Also, the design of the RPS in accordance with IEEE 603-2018 supports compliance with PDC-RFDC 20 because the standard provides the minimum functional and design criteria for SR I&C systems. 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 NRC staff's review of the information provided in the PSAR on how the RPS is designed in accordance with IEEE 603-2018 and the use of ANSI/ISA 67.04.01-2018, the NRC staff finds that the preliminary design of the RPS is consistent with 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 inservice 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 is consistent with 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 DBA 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 that the architecture is configured with independent divisions. As specified in the NRC staff's 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 that the preliminary design of the RPS is consistent with 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 that the preliminary design of the RPS is consistent with 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 NRC 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 that the preliminary design of the RPS is consistent with 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 that the preliminary design of the RPS is consistent with 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 that the preliminary design of the RPS is consistent with 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 DBEs and DBAs can neither: (1) result in damage to the reactor HPB 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 that the preliminary design of the RPS is consistent with 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 I&C, 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 that the preliminary design of the RPS is consistent with PDC-RFDC 30.

Safety Functions

PSAR section 6.4.13.1 states that the RPS is designed to support RSFs 1.1.1, 1.1.2, and 1.3.1. LME identified PDC-RFDC 13, PDC-RFDC 20, PDC-RFDC 26, and PDC-RFDC 30 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 TS 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 to protect 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 SLs and LCOs are met. As stated in part IV of the CP application, information on specific limiting safety system settings (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 that the information provided for the RPS meets the requirements of 10 CFR 50.34(a)(5) because the variables, conditions, and items that are likely to become subjects of TS for the RPS are identified. Consistent with 10 CFR 50.34(b)(6)(vi), the NRC staff will evaluate the proposed TS 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's 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 that 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 NRC staff would verify that the ASAIs have been addressed as part of its review of an OL application.

Codes and Standards - Safety Systems

Paragraph 50.55a(h)(3) of 10 CFR identifies, in part, that CPs 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 NRC 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 DID assessment is encompassed by the overall plant DID 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 10 CFR 50.34(a)(3)(iii), the NRC staff finds that the information provided on how the RPS can meet the requirements of 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 NRC staff's review of a future OL application, the NRC 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's 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 that 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 the requirements of 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 TS; and

- is sufficient to show how the RPS can meet the requirements of 10 CFR 50.55a(h)(3), with a final determination of whether 10 CFR 50.55a(h)(3) is met to be addressed at the OL stage.

6.4.14 Nuclear Instrumentation System

PSAR section 6.4.14 describes the 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 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.

The system-specific regulatory requirements 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 and the correction sheet dated January 3, 1995.

The 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; and
- 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 that 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 the requirements of 10 CFR 50.55a(h)(3). The NRC staff's 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 QA requirements consistent with LME's QAP, as described in PSAR section 8.1. The NRC staff evaluated LME's QAP in SE section 8.1. Based on the NRC staff's evaluation of the information provided in the PSAR, the NRC staff finds that the preliminary design of the NIS is consistent with PDC 1 by using nuclear standards endorsed in NRC regulatory guidance and the LME QAP.

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 to be necessary to protect the safety-significant SSCs and ensure performance of safe-shutdown functions. The fire protection system is part of the plant FPP which is designed to meet the requirements of 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 that 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)(3), and IEEE 7-4.3.2-2016, as endorsed by RG 1.152. The NIS SSCs subject to TS undergo periodic surveillances. The NIS is also subject to preoperational testing and startup testing as described in PSAR section 12.1.1 and evaluated in section 12.1 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 of this SE, and by using nuclear standards endorsed in NRC regulatory guidance, the NRC staff finds that 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 20 and PDC-CDC 26. The NRC staff's evaluation of the specific PDCs that support meeting the requirements 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 20 and PDC-CDC 26, as discussed later in this section of the SE, the NRC staff finds that 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 NRC 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 20 and PDC-RFDC 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 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. Also, the design of the NIS in accordance with IEEE 603-2018 supports the compliance with PDC-RFDC 20 and PDC-CDC 20 because the standard provides the minimum functional and design criteria for SR I&C systems. Based on the NRC staff's review of the information provided in the PSAR, the NRC staff finds that 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 20 and PDC-RFDC 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 that 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 DBA 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 that 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 NRC staff finds that 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 CCF 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 that 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 met 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 that 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 that 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 that 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 that 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 HPB 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 that 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 that 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 that 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 that 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 that the preliminary design of the NIS is consistent with PDC-CDC 34.

Safety Functions

PSAR section 6.4.14.1 states that the NIS is designed to support RSFs 1.1.1, 1.1.2, and 1.3.1. 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, 1.2.2, 1.3.2, and 2.2.1. 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 PSFs 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 and NSRST 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 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 TS 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 to protect 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 SLs 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 that the information provided for the NIS meets the requirements of 10 CFR 50.34(a)(5) because the variables, conditions, and items that are likely to become subjects of TS for the NIS are identified. Consistent with 10 CFR 50.34(b)(6)(vi), the NRC staff will evaluate the proposed TS when detailed design information is available at the OL application stage.

Codes and Standards - Safety Systems

Paragraph 50.55a(h)(3) of 10 CFR identifies, in part, that CPs 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 NRC 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 10 CFR 50.34(a)(3)(iii), the NRC staff finds that the information provided on how the NIS can meet the requirements of 10 CFR 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 NRC staff's review of a future OL application, the NRC staff will confirm that the final design meets the requirements of 10 CFR 50.55a(h)(3) and evaluate acceptability of 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 NRC staff confirmed that these fundamental principles are reflected in the preliminary RPS architecture, which includes the NIS sensors, as shown in PSAR figure 6.4.13-1. Additionally, 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 requirements of 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.

6.4.14.2 Conclusion

Based on the review described above, the NRC staff concludes that the information provided in PSAR section 6.4.14 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 NIS 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 NIS by identifying key design features of the NIS as probable subjects of TS; and

- is sufficient to show how the NIS can meet the requirements of 10 CFR 50.55a(h)(3), with a final determination of whether 10 CFR 50.55a(h)(3) is met to be addressed at the OL stage.

7 NON-SAFETY-RELATED WITH SPECIAL TREATMENT STRUCTURE, SYSTEM, AND COMPONENT CRITERIA AND CAPABILITIES

This chapter of the safety evaluation (SE) summarizes the U.S. Nuclear Regulatory Commission (NRC or Commission) staff's review and evaluation of the Long Mott Generating Station (LMGS) preliminary safety analysis report (PSAR) chapter 7, "Non-Safety-Related with Special Treatment Structure, System, and Component Criteria and Capabilities."

7.1 Reliability and Capability Targets for Non-Safety-Related with Special Treatment Structures, Systems, and Components

PSAR section 7.1 describes the applicant's approach to developing reliability and capability targets for non-safety-related with special treatment (NSRST) structures, systems, and components (SSCs) and presents the set of targets developed based on the preliminary design.

The applicable regulatory requirements for the evaluation of reliability and capability targets for NSRST SSCs are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a)(4); and
- 10 CFR 50.35(a)(1).

The applicable guidance for the review of PSAR section 7.1 is as follows:

- Regulatory Guide (RG) 1.233, Revision 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 Nuclear Energy Institute (NEI) 18-04, Revision 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," (ML19241A472), with clarifications.
- RG 1.253, Revision 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 NEI 21-07, Revision 1, "Technology Inclusive Guidance for Non-Light Water Reactor Safety Analysis Report: For Applicants Utilizing NEI 18-04 Methodology" (ML22060A190), with clarifications and additions.

7.1.1 Technical Evaluation

As described in PSAR section 1.3.1, the applicant is implementing the licensing modernization project (LMP) methodology as outlined in RG 1.233, which endorses 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, "General Plant and Site Description and Overview of Licensing Basis," Long Mott Energy, LLC., (LME) is generally following the content and organization guidance provided in NEI 21-07 as endorsed with clarifications in RG 1.253. NEI 21-07, Revision 1, section C.7.1, "Reliability and Capability Targets for NSRST SSCs," contains applicable guidance for the content and review of PSAR section 7.1.

The NRC staff evaluated PSAR section 7.1 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-design-basis accident (non-DBA) licensing basis events (LBEs) and to inform the selection of special treatments for safety-related (SR) and NSRST SSCs. LME provides preliminary plant-level reliability and capability targets for NSRST SSCs in PSAR table 7.1-1, 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 support the NRC staff's findings for 10 CFR 50.34(a)(4) and 10 CFR 50.35(a)(1).

The preliminary targets provided in the PSAR are limited to NSRST PRA safety functions (PSFs) 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 will 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 7.1-1 defines the preliminary reliability and capability targets for the following 13 unique NSRST PSFs:

- NSRST PSF 1.1.3.1: Control core reactivity to preclude reactor trips
- NSRST PSF 1.2.2.2: Restore active heat removal
    - NSRST PSF 1.2.2.2.1: Restore active core heat removal in the absence of primary helium flow
    - NSRST PSF 1.2.2.2.2: Restore active core heat removal in the absence of primary system pressure
    - NSRST PSF 1.2.2.2.3: Maintain active core heat removal in the absence of secondary heat removal
    - NSRST PSF 1.2.2.2.4: Maintain active core heat removal during events that involve water/steam ingress
- NSRST PSF 1.2.2.3: Control heat removal of the reactor cavity through active means
- NSRST PSF 1.3.3: Control water/steam ingress
- NSRST PSF 1.3.2.1: Remove water/steam from the steam generator
- NSRST PSF 2.2.1: Isolate helium pressure boundary
- NSRST PSF 2.7.1: Control the frequency of small depressurization events

- NSRST PSF 2.7.2: Control the frequency of medium depressurization events
- NSRST PSF 2.7.3: Control the frequency of large depressurization events

Two NSRST PSFs are listed twice in the table, PSF 1.2.2.2 and PSF 1.2.2.2.3, to capture unique capability targets related to achieving the PSF. As the PRA, the safety analysis, and the associated reliability and capability targets continue to develop, these may be assigned distinct PSFs, and the SSCs the PSFs are associated with will be identified and documented. Based on the preliminary nature of the PRA and design, the NRC staff determined that the approach of capturing multiple aspects of a function under one PSF is reasonable for the CP stage.

PSAR table 7.1-1 does not include the functions of some NSRST SSCs: those classified as NSRST based on providing a load path for SR SSCs. This approach to classification is inconsistent with the methodology in NEI 18-04 and LME has provided its justification for this deviation in PSAR section 5.1. LME will determine the reliability and capability targets for these SSCs through its structural and seismic design requirements and the adequacy of these targets to support the overall safety analysis will be confirmed through the seismic PRA, as discussed in PSAR section 3.7 and evaluated in section 3.7.1 of this SE.

PSAR table 7.1-1, defines the reliability targets to be the following three for the majority of the NSRST PSFs listed:

- Maintain frequency of all LBEs with failure of this PSF below 1E-2 per plant year.
- Maintain LBEs with failure of this NSRST PSF within the frequency-consequence (F-C) target.
- Maintain cumulative risk metrics within targets.

However, the reliability targets for NSRST PSFs 1.3.2.1, 2.2.1, 2.7.1, and 2.7.2 are:

- Maintain LBEs with failure of this NSRST PSF within the F-C target.
- Maintain cumulative risk metrics within targets.

During the regulatory audit (ML26293A340), LME stated that, for NSRST PSFs 1.3.2.1, 2.7.1, and 2.7.2 the reliability target, “Maintain frequency of all LBEs with failure of this PSF below 1E-2 per plant year” is not used because the NSRST PSF classification is based on the judgment of the integrated decision-making process panel (IDPP), and that for NSRST PSF 2.2.1, the reliability target is not used because PSF 2.2.1 is classified as NSRST due to risk significance. The NRC staff determined that the information provided is sufficient for the CP based on the preliminary design stage and will review the system-level reliability targets that LME plans to provide in an OL application.

The applicant stated that the basis for the functional-level reliability targets is derived from the development of the frequency targets for a collection of LBEs that are prevented or mitigated by a particular NSRST. However, the reliability targets in PSAR table 7.1-1 align with the overall requirements that the plant should meet under LMP, without further definition of the reliability various functions will need to achieve. Therefore, because the reliability targets align with the

overall 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 and defining a methodology for establishing functional and SSC-level targets at the OL stage is consistent with RG 1.253, which endorses NEI 21-07. The plant-level targets associated with the F-C target and cumulative risk metrics are discussed in PSAR chapter 3, "Methodologies, Analyses, and Licensing Basis Events," and chapter 4, "Integrated Evaluations."

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 proposed a capability target for each PSF identified in PSAR table 7.1-1. 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 NSRST PSF at the functional level and that the capability target 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 NSRST PSFs and associated 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.

As described in PSAR sections 6.3 and 7.2, LME has established a process for assigning reliability and capability targets and special treatments that ensures the full scope of LBEs are addressed within the design of the SSCs. This process is necessary to address the gap that exists between the principal design criteria (PDCs) that LME has established and the LBEs in which the SSCs are assumed to function within the safety analysis. Because the reliability and capability targets and special treatments from this process are necessary to address this gap, the NRC staff considers these principal architectural and engineering criteria. In reaching its reasonable assurance finding, the NRC staff was guided by, among other things, considerations of how the processes described in the PSAR will address any gaps in the PDCs.

7.1.2 Conclusion

Based on its review, the NRC staff finds that the information on the reliability and capability targets for NSRST SSCs, as described in PSAR section 7.1, is consistent with the applicable regulatory guidance and meets the regulatory requirements of10 CFR 50.34(a)(4).

7.2 Special Treatment Requirements for NSRST SSCs

PSAR section 7.2 describes the applicant's approach to selecting special treatment requirements for NSRST SSCs. Special treatment requirements as defined in NEI 18-04 and stated in PSAR section 7.2 are "requirements that provide increased assurance beyond normal industrial practices that SSCs perform their design-basis functions."

The applicable regulatory requirements for the evaluation of NSRST SSCs are as follows:

- 10 CFR 50.34(a)(4); and
- 10 CFR 50.35(a)(1).

The applicable guidance for the review of PSAR section 7.2 is as follows:

- RG 1.233, Revision 0, which endorses NEI 18-04, Revision 1, with clarifications.
- Division of Advanced Reactors and Non-Power Production and Utilization Facilities (DANU) Interim Staff Guidance (ISG) DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications - Roadmap," section 1.1, "SAR Chapters 1-8."
- RG 1.253, Revision 0, which endorses NEI 21-07, Revision 1, with clarification and additions.

7.2.1 Technical Evaluation

RG 1.253, which endorses NEI 21-07 with clarifications, provides applicable guidance in section C.7.2, "Special Treatment Requirements for NSRST SSCs" for the content and review of PSAR section 7.2.

PSAR section 7.2 states that the systematic process that LME uses for special treatment selection will be presented in the final safety analysis report (FSAR). The systematic process for selecting special treatments for NSRST SSCs considers the 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 the regulatory audit. During the regulatory audit, the NRC staff and LME engaged in a discussion of how this process is implemented by LME. The NRC staff confirmed that the preliminary process is being used to systematically evaluate the special treatments being applied to each system with NSRST components. The process also includes an IDPP review of design documents prior to a review meeting that includes representatives from safety, engineering, operations, maintenance, etc.

A preliminary set of special treatments are described in the SSC descriptions within PSAR section 7.3. LME has selected nuclear codes and standards as special treatments for the design of some NSRST SSCS. The use of nuclear codes and standards is considered a special treatment in the NEI 18-04 methodology. Additionally, application of the programs (e.g., quality

assurance (QA), reliability integrity management, inservice testing (IST)) described in PSAR chapter 8, “Plant Programs,” are considered special treatments in NEI 18-04. Section 7.3 of this SE, which includes technical evaluation sections for the NSRST systems and structures, notes where LME applied nuclear codes and standards, design or construction requirements beyond commercial, or programmatic special treatments as special treatments. Special treatments for NSRST portions of SR systems are noted in section 6.4 of this SE.

As stated in PSAR section 7.2, 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 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, complementary design criteria (CDCs), required functional design criteria (RFDCs), and safety-related design criteria (SRDCs), as discussed in section 5.3 of this SE. Because the special treatments are necessary to address the gap caused by the limited scope, the NRC staff considers these principal architectural and engineering criteria. In reaching its reasonable assurance finding, the NRC staff was guided by, among other things, considerations of how the processes described in the PSAR will address any gaps in the PDCs, CDCs, RFDCs, and SRDCs.

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

The NRC staff reviewed PSAR section 7.2 against the guidance in NEI 21-07. The preliminary special treatments described in the subsections of PSAR section 7.3 are sufficient for the CP stage because they ensure that the systems can perform the NSRST PSFs for which they are being relied on, consistent with the preliminary state of the design. The format and content presented in PSAR table 7.2-1 is consistent with table 7.1 of NEI 21-07 and provides a reasonable approach to summarize the special treatments in the FSAR.

7.2.2 Conclusion

The NRC staff reviewed the approach to assigning special treatment requirements to the NSRST SSCs described in PSAR section 7.2 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 clarification by RG 1.233. The PSAR description is sufficient to address the methodology portions of the associated regulatory requirements of 10 CFR 50.34(a)(4).

The PSAR content is sufficient for the CP stage based on the special treatments being described as part of the preliminary design and the format and content proposed for section 7.2 of the FSAR being reasonable and consistent with the guidance in NEI 21-07, as endorsed with clarifications and additions in RG 1.253.

7.3 Descriptions for NSRST SSCs

PSAR section 7.3 provides descriptions for the NSRST SSCs. Each subsection in PSAR section 7.3 generally includes the following sections:

- Functions and design criteria: lists NSRST PSFs, lists associated PDCs, provides the system-specific 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, which includes key design requirements.

- Qualification, testing, and programs: describes the planned qualification and any testing to be performed, including first-of-a-kind (FOAK) testing and describes the preliminary special treatments selected, including programmatic special treatments, for the system.

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

The applicable regulatory requirements for the design of SR and NSRST SSCs evaluated in sections 7.3.1 through 7.3.12 of this SE are as follows:

- 10 CFR 50.34(a)(3);

- 10 CFR 50.34(a)(4); and

- 10 CFR 50.35, “Issuance of Construction Permits.”

The system-specific regulatory requirements will be listed, as applicable, in the subsections containing the system-specific evaluation.

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

- RG 1.233, Revision 0, which endorses, NEI 18-04, Revision 1, with clarifications.

- RG 1.253, Revision 0, which endorses NEI 21-07, Revision 1, with clarifications and additions.

- DANU-ISG-2022-01, “Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications - Roadmap”

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

PDCs applicable to all NSRST systems 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

The 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 7.3 and the NRC staff's evaluations provided in each of the subsections below, the NRC staff finds that the applicant's identification of applicable PDCs is acceptable. The NRC staff notes that the discussions pertaining to PDC 5 below refer to "modules" or "units." The language in the approved PDC 5 for the Xe-100 design refers to "units;" however, for LMGS, "modules" are equivalent to "units" for the purpose of evaluating PDC 5.

As discussed in section 5.3 of this SE with respect to PDCs and section 5.2 of this SE with respect to CDCs, LME has limited the scope of PDCs and CDCs to anticipated operational occurrences (AOOs) in a way that is inconsistent with the LBEs relying on the NSRST SSCs. LME is addressing the safety impacts of this inconsistency 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 PSF, as discussed in PSAR section 6.3. LME did not include the system-specific reliability targets and SSC-specific capability targets in the CP application. The NRC staff determined that the preliminary special treatments for the SSCs support ensuring the safety functions relied on, can be performed adequately for the full scope of LBEs. LME will provide the final PDCs that will be applied to each safety-significant SSC at the OL application stage. The NRC staff will evaluate the PDCs to determine if inconsistencies remain and will confirm that the safety aspects are addressed appropriately through the final reliability and capability targets and special treatments applied to the SSCs.

7.3.1 Civil Structures

PSAR section 7.3.1 describes the civil structures which include the following NSRST nuclear island (NI) structures:

- Fuel handling annex building (FHAB)
- Inter-unit access tunnel (IUAT)
- Spent fuel intermediate storage facility (SFISF)
- Canister processing facility (CPF)
- Nuclear island auxiliary structure (NIAS)
- Controls and electrical building (CEB)

The system-specific regulatory requirements for the civil structures are as follows:

- 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants."

The system-specific guidance for the evaluation of the civil structures is as follows:

- 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).

In addition to PDCs 1-6, the following PDCs are applicable to the SFISF, the spent fuel canisters (SFCs) and storage racks as described below:

- PDC 61 - Fuel storage and handling and radioactivity control
- PDC 62 - Prevention of criticality in fuel storage and handling
- PDC 63 - Monitoring Fuel and Waste Storage

7.3.1.1 Technical Evaluation

The NRC staff reviewed PSAR section 7.3.1, which describes NSRST structures identified via the integrated decision-making process (IDP). These NSRST structures include:

- Fuel handling annex building (FHAB)
- Inter-unit access tunnel (IUAT)
- Spent fuel intermediate storage facility (SFISF)
- Canister processing facility (CPF)
- Nuclear island auxiliary structure (NIAS)
- Controls and electrical building (CEB)

For each structure, the NRC staff evaluated its safety significance, compliance with applicable PDCs, and safety functions.

7.3.1.1.1 Fuel Handling Annex Building

The FHAB consists primarily of a structural steel superstructure with a portion of the FHAB adjacent to the shield structure (SST) (i.e., the entry points into the SST) consisting of precast concrete walls required for radiation shielding. PSAR figure 7.3.1-1 shows, in part, an elevation view of the FHAB, showing the structural steel portion of the FHAB and a precast portion of the FHAB. The figure also shows top of slab elevations for the multiple floor/roof elevations of the FHAB**.** Additionally, PSAR figures 7.3.1-2 through 7.3.1-5 show plan views and related dimensions for multiple elevations of the FHAB. PSAR section 3.7 describes, in part, the analysis and design criteria and materials applicable to the FHAB.

Safety Significance

As described in PSAR section 7.3.1.1, the FHAB provides operational and maintenance space for the mechanical and electrical systems located within the building and is used to store, handle, and load fresh fuel into the fuel handling system (FHS). The building houses the FHS, including the remotely operated special tooling (ROST), reactivity control and shutdown system (RCSS), and reactor protection system (RPS) cabinet rooms, among others. Further, the

building serves as the main entry point into the reactor bays. PSAR section 5.1.1 addresses the safety classification criteria and the associated basis applicable to the FHAB including the implementation of design and analysis special treatments that achieve performance levels consistent with the SSCs being supported/protected by the FHAB. Moreover, per PSAR section 5.5.2 and table 5.5-21, while not modeled in the PRA at the CP stage, the FHAB is one of the SSCs that was classified as NSRST for DID adequacy via the IDP. PSAR table 5.5-21 and section 7.3.1.1.1 indicate that the PSF for the FHAB is to prevent excessive challenges to the fuel and other SR SSCs within the building in response to external hazards due to building collapse. SE section 5.1.1 provides additional details on the NRC staff's evaluation of the safety classification.

Compliance with Applicable PDCs

PSAR section 7.3.1.1.1 indicates that PDC 1 through 6 are applied to the design of safety-significant SSCs, including the FHAB.

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR table 7.3.1-1, which provides the list of codes and standards applicable to the preliminary structural design of the FHAB. The design basis for the FHAB includes the use of commercial standards including American National Standards Institute (ANSI)/American Institute of Steel Construction (AISC) ANSI/AISC 360 and American Concrete Institute (ACI) 318 for the structural steel and concrete portions of the building, respectively, with enhancements applied as special treatments that include a conservative seismic demand and use of provisions from nuclear standards such as American Society of Civil Engineers (ASCE) 4, ASCE 43, ANSI/AISC N690, and ACI 349 among other codes and standards listed in PSAR table 7.3.1-1. Examples of supplemental provisions from nuclear codes and standards used in the design of the FHAB include analyzing the FHAB in accordance with the provisions of ASCE 43 and ASCE 4, the use of **[[**

]], and the use of provisions from ANSI/AISC N690 and ACI 349 for impulsive and impactive loads such as hurricane/tornado missiles and high-energy line break (HELB) loads, among others.

The NRC staff also reviewed the PSAR for QA treatment applicable to the FHAB. PSAR section 7.3.1 indicates the applicability of the LMGS quality assurance program description (QAPD), described in PSAR section 8.1, to the FHAB. The FHAB is designed, fabricated, constructed, and tested in accordance with the quality assurance program (QAP) described in part III of the LMGS QAPD and PSAR section 8.1. Additional special treatments related to QA will be specified at the OL application stage. The application of part III of the LMGS QAPD at the CP stage is sufficient as a preliminary special treatment to meet the QA requirements under PDC 1 for the FHAB. The NRC staff's evaluation of the LMGS QAPD is provided in the QAPD topical report (TR) SE and section 8.1 of this SE.

The NRC staff finds that LME's preliminary design bases for the FHAB are consistent with PDC 1 because its 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 FHAB is within the scope of LME's QAP.

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 FHAB is categorized as a seismic category IIA (SC-IIA) structure and is designed to withstand an SC-IIA event, namely the SC-IIA design response spectrum (DRS) described in PSAR section 3.7.3.3, which is equivalent to the **[[ ]]** The SC-IIA DRS is conservative relative to the site-specific ground motion response spectrum at the LMGS site. Further, per PSAR section 7.3.1.1.1, the FHAB is designed in accordance with the seismic design basis for NSRST structures detailed in PSAR section 3.7. 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 the consideration of soil structure interaction (SSI) effects are evaluated in section 3.7 of this SE. **[[ ]]** Based on LME's use of seismic design criteria that is consistent with the analysis and design criteria for SC-I structures, including the **[[ ]]** the NRC staff finds that LME's approach for protecting the FHAB is acceptable, is consistent with PDC 2, and meets the requirements of 10 CFR Part 50, Appendix S with respect to the consideration of the effects from earthquakes.

The NRC staff reviewed LME's consideration of tornado and hurricane effects on the FHAB. Per PSAR sections 3.7 and 7.3.1.1.1, the FHAB is designed to withstand tornado and hurricane forces and missile impacts. Additional considerations of tornado and hurricane effects are addressed in PSAR sections 3.7.4.10 and 3.7.4.11, and PSAR table 7.3.1-1. Per these PSAR sections and table, the tornado and hurricane forces and missiles impacts are considered in the design of the FHAB in accordance with RG 1.76, Revision 1; RG 1.221; ANSI/AISC N690; and ACI 349, which are applicable to SR structures. Based on LME's use of RG 1.76, RG 1.221, ANSI/AISC N690, and ACI 349-13 in its consideration of tornado and hurricane effects on the FHAB, the NRC staff finds that LME's approach for protecting the FHAB from the tornado and hurricane hazards is acceptable and is consistent with PDC 2.

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. PSAR 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 7.3.1.1.1, the structural integrity of the FHAB under design-basis hazard level (DBHL) flooding conditions is ensured by considering flood loads in the design of the structure as described in PSAR section 3.7. PSAR section 7.3.1.1.1 also states that the design of the FHAB accounts for the flooding DBHL to protect SR SSCs located within the FHAB by: (1) preventing water intrusion with openings that are either located above the flood height, designed to be substantially impermeable, or designed with an equivalent preventative measure; (2) elevating SR SSCs above the flood height where possible; and (3) designing SR SSCs that cannot be elevated such that their ability to fulfill a required safety function (RSF) is not impacted if submerged. For example, the SR RPS and RCSS cabinets are raised above the external flood stillwater height on a mezzanine level that is accessible to operators. 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 7.3.1-1, the NRC staff finds that LME's approach for protecting the FHAB from the flooding hazard is acceptable and is consistent with PDC 2. Further, the NRC staff determined that the strategies

for protecting the SR SSCs located within the FHAB are adequate and are consistent with PDC 2 because they either eliminate the flooding DBHL hazard for such SSCs or ensure appropriate qualification of the SSCs under the flooding DBHL.

The NRC staff also reviewed PSAR section 3.7, which describes the loads and load combinations applicable to NSRST structures, including the FHAB. Per the design criteria in PSAR section 3.7, in addition to considering the effects of natural phenomena, the design of the FHAB 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 loads considered include normal, severe environmental, extreme environmental, and abnormal loads. The NRC staff finds that the loads and load combinations described in PSAR section 3.7 are consistent with 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 further evaluates loads and load combinations for LMGS structures 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 FHAB to protect it against natural phenomena are consistent with PDC 2.

Fire Protection (PDC 3)

The NRC staff reviewed PSAR section 7.3.1, which describes that the FHAB is designed and fabricated using materials that are high-temperature-resistant, inherently noncombustible, and non-fire sustaining. Moreover, in general, this section describes that the probability and effects of fires in the FHAB due to fixed and transient combustible materials, is minimized by the implementation of administrative controls. In addition, as indicated in PSAR section 7.3.1, LME's fire protection strategies include the implementation of fire protection systems that are designed, installed, and tested in accordance with National Fire Protection Association (NFPA) standards. The NRC staff reviewed LME's approach for meeting PDC 3 and finds 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 10 CFR 50.48(a) and PDC 3 by minimizing the probability and effects of fires and explosions. These elements include, in part, implementation of engineered fire detection and suppression systems to control fires; and 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 7.3.1 and 8.6, as it relates to the FHAB, and determined that the preliminary design bases for the FHAB are consistent with PDC 3. This conclusion is based on LME's use of construction materials that are high-temperature-resistant, inherently noncombustible, and non-fire sustaining, and LME's commitment to implementing fire protection systems and features as necessary, consistent with its FPP. The NRC staff will review fire zones and combustible loading along with fire protection systems and features related to the FHAB at the OL application stage. LME's approach for meeting the requirements of PDC 3 and the NRC staff's evaluation described above are also applicable to the SFISF, IUAT, CPF, NIAS, and CEB.

Environmental and Dynamic Effects Design Bases (PDC 4)

The NRC staff reviewed PSAR section 7.3.1 and the other PSAR sections, mentioned below, that address how environmental conditions are considered in the design of the FHAB. PSAR sections 3.7.5.1 and 3.7.5.2 summarize, in part, the applicable loads and load combinations for NSRST concrete and steel structures, respectively, including the FHAB. These loads include normal, severe environmental, extreme environmental, and abnormal loads, including HELB loads. PSAR sections 6.1.1.7 and 3.7.4.14 describe the characterization of HELB loads and postulated HELB sources. Further, PSAR section 7.3.1 indicates that the NSRST structures outside the reactor building (RB), including the FHAB, 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 NSRST structures, including the FHAB, is consistent with the provisions of ACI 349-13 and ANSI/AISC N690, as endorsed by RGs 1.142 and 1.243, respectively, and is therefore acceptable. Additionally, PSAR section 7.3.1 indicates that the NSRST structures, including the FHAB, 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 7.3.1-1, these practices include provisions from AISC 360, AISC N690, ACI 318, and ACI 349 for protecting steel and concrete structures against corrosion effects. Similarly, these practices 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. Additional discussion of fire protection is addressed under the evaluation of PDC 3.

Based on the NRC staff's evaluation described above, the NRC staff finds that LME's preliminary design bases for the FHAB are consistent with PDC 4.

Sharing of Structures, Systems, and Components (PDC 5)

The NRC staff reviewed PSAR sections 1.1.3 and 7.3.1, which cover information, in part, related to the sharing of the FHAB. Per PSAR section 7.3.1.1.1, the FHAB acts as the main entry point for the reactor citadels, providing room for equipment required for a radiological entry point, such as the radiation protection desk. Further, the FHAB houses the FHS and related equipment and, per PSAR section 1.1.3, all SR SSCs in the FHAB are compartmentalized by module. PSAR section 7.3.1.1.2 describes that the FHAB is designed to protect equipment inside the FHAB and, per PSAR section 7.3.1.1.1, the FHAB is designed to the NSRST criteria in PSAR section 3.7. The NRC staff's evaluation of the NSRST design criteria applicable to the FHAB is addressed in SE section 3.7 and above under the evaluation of how the FHAB is consistent with PDCs 1, 2, and 4.

Based on the design criteria for the FHAB that is focused on protecting the SSCs within the FHAB, the NRC staff concludes that the sharing of the FHAB is adequately considered in the design criteria for the FHAB, and therefore, the NRC staff finds that LME's preliminary design bases for the FHAB are consistent with PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR section 7.3.1, which addresses LME's consideration of monitoring, inspection, testing, and surveillance criteria as part of the design bases for the

FHAB. Per this PSAR section, these structures are designed to permit appropriate monitoring, periodic inspection, testing, and/or surveillance. These activities are performed in accordance with the maintenance program described in PSAR section 8.3 and include visual examinations and structural integrity/health monitoring through temperature, seismic, and radiation measurements, among others. Moreover, per PSAR section 8.3, these activities are informed by the guidance in Nuclear Management and Resources Council (NUMARC) 93-01, Revision 4F, as endorsed by RG 1.160, and additional details will be provided at the OL application stage. The NRC staff's evaluation of PSAR section 8.3 is in section 8.3 of this SE. The NRC staff's review determined that LME's approach described above, is informed by the guidance in NUMARC 93-01 as endorsed by RG 1.160. On this basis, the NRC staff concludes that LME's preliminary design of the reactor support structures and common SR foundation is consistent with PDC 6 and that the review of detailed maintenance program requirements associated with the final design of structures can be reasonably addressed at the OL application stage. LME's approach for meeting the requirements of PDC 6 and the NRC staff's evaluation described above are also applicable to the, IUAT, SFISF, CPF, NIAS, CEB.

Safety Functions

PSAR section 7.3.1.1.1 states that the FHAB is classified as NSRST for preventing excessive challenges to the fuel and other SR SSCs within the building in response to external hazards due to building collapse. As discussed under PDCs 1, 2, and 4 above, LME's design approach for the FHAB takes into consideration the postulated loads from normal, severe environmental, extreme environmental, abnormal conditions, and associated load combinations in accordance with the applicable codes and standards. Further, LME's approach implements enhancements over applicable commercial standards, applied as special treatments, that include the use of the SC-IIA design-basis earthquake, the consideration of extreme hurricane/tornado loads and HELB loads consistent with the criteria for SR structures, and supplementing the commercial codes and standards with provisions from nuclear codes and standards. Additionally, **[[**

]] Based on the above, the NRC staff finds that LME's design bases for the FHAB will achieve adequate structural integrity to support performance of the FHAB's PSF of preventing excessive challenges to the fuel and other SR SSCs within the FHAB.

7.3.1.1.2 Inter-Unit Access Tunnel

As described in PSAR section 7.3.1.2.1 and PSAR figures 7.3.1-7 and 7.3.1-8, the IUAT is a one-level reinforced concrete structure. PSAR figures 7.3.1-7 and 7.3.1-8 show elevation views for the IUAT and associated dimensions**.** PSAR figure 7.3.1-6 shows the plan view of the IUAT and associated dimensions. PSAR section 3.7 describes, in part, the analysis and design criteria and materials applicable to the IUAT.

The NRC staff notes that PSAR sections 1.1.3 and 7.3.1.2.1 mention the construction of a total of four SFISFs. Further, PSAR section 7.3.1.2.1 states that when entirely constructed, the IUAT will branch out to all four SFISFs. The NRC staff's review of the IUAT focuses on the IUAT as described in PSAR section 3.7.1.2 and depicted in PSAR figures 7.3.1-6 through 7.3.1-8 (i.e., a IUAT structure branching out to a single SFISF).

Safety Significance

As described in PSAR section 7.3.1.2.1, the IUAT provides radiation shielding and a transportation corridor for the movement of the ROST carts and the respective SFCs they carry from the CPF to the SFISF.

PSAR section 5.1.1 addresses the safety classification criteria and associated basis applicable to the IUAT including the implementation of design and analysis special treatments that achieve performance levels consistent with the SSCs being supported or protected by the IUAT. Moreover, per PSAR section 5.5.2 and table 5.5-21, while not modeled in the PRA at the CP stage, the IUAT is one of the SSCs that was classified as NSRST for DID adequacy via the IDP. PSAR table 5.5-21 and section 7.3.1.2.1 indicate that the PSF for the IUAT is to prevent mechanical challenges to the fuel in response to external hazards due to building collapse. Additional details on the NRC staff's review of safety classification are in section 5.1.1 of this SE.

Compliance with Applicable PDCs

PSAR section 7.3.1.2.1 states that PDC 1 through 6 are applied to the design of safety-significant SSCs, including the IUAT.

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR table 7.3.1-2, which provides the list of codes and standards applicable to the preliminary structural design of the IUAT. Like for the FHAB, the design criteria for the NSRST IUAT includes the use of commercial codes and standards with enhancements applied as special treatments that include a conservative seismic demand and the use of provisions from nuclear codes and standards including ASCE 4, ASCE 43, and ACI 349-13 among other codes and standards listed in PSAR table 7.3.1-2.

The NRC staff also reviewed the PSAR for QA treatment applicable to the IUAT. PSAR section 7.3.1 indicates the applicability of the LMGS QAPD, described in PSAR section 8.1, to the IUAT. The IUAT is designed, fabricated, constructed, and tested in accordance with the QAP described in part III of the LMGS QAPD and PSAR section 8.1. Additional special treatments related to QA will be specified at the OL application stage. The application of part III of the LMGS QAPD at the CP stage is sufficient as a preliminary special treatment to meet the QA requirements under PDC 1 for the IUAT. The NRC staff's evaluation of the LMGS QAPD is provided in the QAPD TR SE and section 8.1 of this SE.

The NRC staff finds that LME's preliminary design bases for the IUAT are consistent with PDC 1 because its design bases includes the use of well-established codes and standards with conservative determination of demands (e.g., seismic demands), the commercial codes and standards used are supplemented as necessary with provisions from nuclear codes and standards, and the IUAT is within the scope of LME's QAP.

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 IUAT is categorized as an SC-IIA structure and is
[[

]] Further, per PSAR section 7.3.1.2.1, the IUAT is designed in accordance with the seismic design bases for NSRST structures detailed in PSAR section 3.7. 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. **[**

] and LME's consideration of SSI effects as described in PSAR section 3.7, the NRC staff finds that LME's approach for protecting the IUAT is acceptable, is consistent with PDC 2 and meets the requirements of 10 CFR Part 50, Appendix S, with respect to the consideration of the effects from earthquakes.

The NRC staff reviewed LME's consideration of tornado and hurricane effects on the IUAT. Per PSAR sections 3.7 and 7.3.1.2.1, the IUAT is designed to withstand tornado and hurricane forces and missile impacts. Additional consideration of tornado and hurricane effects is addressed in PSAR sections 3.7.4.10 and 3.7.4.11 and PSAR table 7.3.1-2. Per these PSAR sections and table, the tornado and hurricane forces and missile impacts are considered in the design of the IUAT in accordance with RG 1.76, Revision 1; RG 1.221; and ACI 349, which are applicable to SR structures. Based on LME's use of RG 1.76, RG 1.221, and ACI 349-13 in its consideration of tornado and hurricane effects on the IUAT, the NRC staff finds that LME's approach for protecting the IUAT from the tornado and hurricane hazards is acceptable and is consistent with PDC 2.

Regarding the consideration of the 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 the consideration of hurricane, tsunami, and seiche hazards, among others. Further, per PSAR section 7.3.1.2.1, structural integrity of the IUAT under DBHL flooding conditions is ensured by considering flood loads in the design of the structure as described in PSAR 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 7.3.1-2, the NRC staff finds that LME's approach for protecting the IUAT from the flooding hazard is acceptable and is consistent with PDC 2.

The NRC staff reviewed PSAR section 3.7, which describes the loads and load combinations applicable to NSRST structures, including the IUAT. Per the design criteria in PSAR section 3.7, in addition to considering the effects of natural phenomena, the design of the IUAT 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 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 are consistent with 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 loads and load combinations for LMGS structures is addressed 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 IUAT to protect against natural phenomena are consistent with PDC 2.

Fire Protection (PDC 3)

LME's approach for meeting the requirements of PDC 3 and the NRC staff's evaluation described above for the FHAB, are also applicable to the IUAT, SFISF, CPF, NIAS, CEB. On this basis, the NRC staff finds that the preliminary design bases for the IUAT are consistent with the requirements of PDC 3.

Environmental and Dynamic Effects Design Bases (PDC 4)

The NRC staff reviewed PSAR section 7.3.1 and the other PSAR sections mentioned below, which address how environmental conditions are considered in the design of the IUAT. PSAR sections 3.7.4 and 3.7.5.2 describe the loads and load combinations used in the design of NSRST concrete structures, including the IUAT. These loads include normal, severe environmental, extreme environmental, and abnormal loads, as applicable. PSAR section 7.3.1 indicates that NSRST structures, including the IUAT, 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 7.3.1 indicates that NSRST structures, including the IUAT, 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. The NRC staff's review determined that these environmental effects as well as the design loads, load combinations, material limit states, and deflection values are in accordance with the applicable design codes and standards as discussed in SE section 3.7 and the evaluation of how the design bases for the IUAT are consistent with PDCs 1 and 2.

Based on the NRC staff's evaluation described above, the NRC staff finds that LME's preliminary design bases for the IUAT are consistent with PDC 4.

Sharing of Structures, Systems, and Components (PDC 5)

The NRC staff reviewed PSAR sections 1.1.3 and 7.3.1, which cover information, in part, related to the sharing of the IUAT. PSAR section 7.3.1.2.1 describes that the IUAT structure connects to the CPF and the SFISF and provides radiation shielding and a transportation corridor for the SFCs from the CPF to the SFISF for long term storage. PSAR section 1.1.3 describes that the IUAT is shared across the modules for these purposes. PSAR section 7.3.1.2.2 describes that the IUAT is designed to protect equipment inside the IUAT and, per PSAR section 7.3.1.2.1, is designed to the NSRST criteria in PSAR section 3.7. The NRC staff's evaluation of the NSRST design criteria applicable to the IUAT, is addressed in SE section 3.7 and above under the evaluation of how the IUAT is consistent with PDC 1, 2, and 4.

Based on the design criteria for the IUAT that is focused on protecting the SSCs within the IUAT, the NRC staff concludes that the sharing of the IUAT is adequately considered in the design criteria for the IUAT, and therefore the NRC staff finds that LME's preliminary design bases for the IUAT are consistent with PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

LME's approach for meeting the requirements of PDC 6 and the NRC staff's evaluation described above for the FHAB, are also applicable to the IUAT, SFISF, CPF, NIAS, CEB. On this basis, the NRC staff finds that the preliminary IUAT is consistent with PDC 6.

Safety Function

Per PSAR section 7.3.1.2.1, the IUAT is classified as NSRST for preventing excessive challenges to the fuel in response to external hazards due to building collapse.

As discussed under PDC 1, 2, and 4 above, the design approach for the IUAT implements commercial standards with enhancements applied as special treatments. These enhancements include: the use of the SC-IIA design-basis earthquake and supplementing the commercial codes and standards with provisions from nuclear codes and standards. Based on the above, the NRC staff finds that LME's design bases for the IUAT will achieve adequate structural integrity to support performance of its PSF.

7.3.1.1.3 Spent Fuel Intermediate Storage Facility

The applicant provided separate descriptions of the design bases for the reinforced concrete structure of the SFISF in PSAR sections 7.3.1.3.1 through 7.3.1.3.3, and the internal support structures for fuel storage in PSAR section 7.3.1.3.4. Accordingly, the NRC staff has separated the assessment here into the concrete structure and the SFCs and storage racks.

Concrete Structure

As described in PSAR section 7.3.1.3.1, the SFISF is a one-level cast-in-place reinforced concrete structure. The concrete roof is cast over a steel plate supported by an internal steel frame structure. PSAR figures 7.3.1-9 and 7.3.1-10 show a plan view and elevation view and associated dimensions for the SFISF, respectively. PSAR section 3.7 describes, in part, the analysis and design criteria and materials applicable to the SFISF. PSAR section 7.3.1.3.1 also provides additional details related to the materials used in the SFISF, consistent with the information in PSAR section 3.7.

Safety Significance

As described in PSAR sections 7.3.1.3 and 7.3.1.3.1, the SFISF includes the physical structure that encompasses the spent fuel storage racks (SFSRs) and associated support frames and the SFISF provides operational space for the ROST carts. The SFISF provides a dedicated space for the intermediate storage of spent fuel throughout the plant operating life and shields plant personnel from ionizing radiation dose outside the perimeter of the structure. PSAR section 3.10.4 describes that the building is also designed to provide reliable passive cooling of the fuel to prevent thermal challenges to the stored fuel that could lead to radionuclide release and, PSAR section 3.9 describes that the building is designed to maintain the storage canisters in an array demonstrated to be safely subcritical. The tri-structural isotropic (TRISO) fuel structure itself provides confinement of radionuclides at limiting fuel temperatures associated with loss of SFISF inlet air flow.

PSAR section 5.1.1 addresses the safety classification criteria and associated bases applicable to the SFISF including the implementation of design and analysis special treatments that achieve performance levels consistent with the SSCs being supported or protected by the SFISF. Per PSAR section 5.5.2 and table 5.5-21, while not modeled in the PRA at the CP stage, the SFISF is one of the SSCs that was classified as NSRST for DID adequacy via the IDP. PSAR table 5.5-21 indicates that the PSF for the SFISF is to prevent excessive challenges to

the fuel in response to external hazards due to building collapse. Additional details on the NRC staff's review of safety classifications are in section 5.1.1 of this SE.

Compliance with Applicable PDCs

PSAR section 7.3.1.3.1 states that PDC 1 through 6 are applied to the design of safety-significant SSCs, including the SFISF. Additionally, per the discussion in PSAR section 7.3.1.3.4, PDC 61 is applicable, in part, to the concrete structure of the SFISF.

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR table 7.3.1-3, which provides the list of codes and standards applicable to the preliminary structural design of the SFISF. The design criteria for the NSRST SFISF include the use of commercial codes and standards with enhancements applied as special treatments that include a conservative seismic demand and the use of provisions from nuclear codes including ASCE 4, ASCE 43, and ACI 349-13, among other codes and standards listed in PSAR table 7.3.1-3.

The NRC staff also reviewed the PSAR for QA treatment applicable to the SFISF. PSAR section 7.3.1 indicates the applicability of the LMGS QAPD, described in PSAR section 8.1, to the SFISF. The SFISF is designed, fabricated, constructed, and tested in accordance with the QAP described in part III of the LMGS QAPD and PSAR section 8.1. Additional special treatments related to QA will be specified at the OL application stage. The application of part III of the LMGS QAPD at the CP stage is sufficient as a preliminary special treatment to meet the QA requirements under PDC 1 for the SFISF. The NRC staff's evaluation of the LMGS QAPD is provided in the QAPD TR SE and section 8.1 of this SE.

The NRC staff finds that LME's preliminary design bases for the SFISF are consistent with PDC 1 because its 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 SFISF is within the scope of LME's QAP.

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 SFISF is categorized as an SC-IIA structure and is designed to withstand an SC-IIA event, **[**

]] Further, per PSAR section 7.3.1.3.1, the SFISF is designed in accordance with the seismic design basis for NSRST structures detailed in PSAR section 3.7. 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 the consideration of SSI effects, are evaluated in section 3.7 of this SE. **[[**

]] Based on LME's use of seismic design criteria that is consistent with the criteria for SC-I structures, including the **[[**

]] and LME's consideration of the SSI effects as described in PSAR section 3.7, the NRC staff finds that LME's approach for protecting the SFISF is acceptable, is consistent with PDC 2, and meets the requirements of 10 CFR Part 50, Appendix S with respect to the consideration of the effects from earthquakes.

The NRC staff reviewed LME's consideration of tornado and hurricane effects on the SFISF. Per PSAR sections 3.7 and 7.3.1.3.1, the SFISF is designed to withstand tornado and hurricane forces and missile impacts. Additional consideration of tornado and hurricane effects is addressed in PSAR sections 3.7.4.10 and 3.7.4.11 and PSAR table 7.3.1-3. Per these PSAR sections and table, the tornado and hurricane forces and missile impacts are considered in the design of the SFISF in accordance with RG 1.76, Revision 1; RG 1.221; and ACI 349, which are applicable to SR structures. Based on LME's use of RG 1.76, RG 1.221, and ACI 349-13 in its consideration of tornado and hurricane effects on the SFISF, the NRC staff finds that LME's approach for protecting the SFISF from tornado and hurricane hazards is acceptable and is consistent with PDC 2.

Regarding the consideration of the 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 the consideration of hurricane, tsunami, and seiche hazards, among others. Further, per PSAR section 7.3.1.3.1, structural integrity of the SFISF under DBHL flooding conditions is ensured by considering flood loads in the design of the structure as described in PSAR 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 7.3.1-3, the NRC staff finds that LME's approach for protecting the SFISF from the flooding hazard is acceptable and is consistent with PDC 2.

The NRC staff reviewed PSAR section 3.7, which describes the loads and load combinations applicable to NSRST structures, including the SFISF. Per the design criteria in PSAR section 3.7, in addition to considering the effects of natural phenomena, the design of the SFISF considers other effects and loads, as described in PSAR sections 3.7.4 and 3.7.5, which include 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 loads and load combinations described in PSAR section 3.7 are consistent with 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 loads and load combinations for LMGS structures is addressed in section 3.7 of this SE.

Based on LME's approach for considering the effects of natural phenomena along with the effects from other loads as described above, the NRC staff finds that LME's preliminary design bases for the SFISF to protect against natural phenomena are consistent with PDC 2.

Fire Protection (PDC 3)

LME's approach for meeting the requirements of PDC 3 and the NRC staff's evaluation described above for the FHAB, are also applicable to the IUAT, SFISF, CPF, NIAS, CEB. On this basis, the NRC staff finds that the preliminary design bases for the SFISF are consistent with PDC 3.

Environmental and Dynamic Effects Design Bases (PDC 4)

The NRC staff reviewed PSAR section 7.3.1 and the other PSAR sections mentioned below that address how environmental conditions are considered in the design of the SFISF. PSAR sections 3.7.4 and 3.7.5.2 describe the loads and load combinations used in the design of NSRST concrete structures, including the SFISF. These loads include normal, severe

environmental, extreme environmental, and abnormal loads, as applicable. PSAR section 7.3.1 indicates that NSRST structures, including the SFISF, 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 7.3.1 indicates that NSRST structures, including the SFISF, 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. The NRC staff's review determined that these environmental effects as well as the design loads, load combinations, material limit states, and deflection values are in accordance with the applicable design codes and standards as discussed in SE section 3.7 and the evaluation of how the design bases for the SFISF are consistent with PDCs 1 and 2.

Based on its evaluation described above, the NRC staff finds that LME's preliminary design bases for the SFISF are consistent with PDC 4.

Sharing of Structures, Systems, and Components (PDC 5)

The NRC staff reviewed PSAR sections 1.1.3 and 7.3.1, which cover information, in part, related to the sharing of the SFISF. PSAR section 7.3.1.3 describes that the SFISF structure encompasses the SFSRs and associated support frames and provides operational space for the ROST carts within the building. The structure serves as a common structure for the storage of spent fuel and protects both the spent fuel and equipment inside the building. Per PSAR section 7.3.1.3.1, the SFISF is designed to the NSRST criteria in PSAR section 3.7. The NRC staff's evaluation of the NSRST design criteria applicable to the SFISF, is addressed in SE section 3.7 and above under the evaluation of how the SFISF is consistent with PDCs 1, 2, and 4.

Based on the design criteria for the SFISF that is focused on protecting the fuel and SSCs within the SFISF, the NRC staff concludes that the sharing of the SFISF is adequately considered in the design criteria for the SFISF, and therefore the NRC staff finds that LME's preliminary design bases for the SFISF are consistent with PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

LME's approach for meeting the requirements of PDC 6 and the NRC staff's evaluation described above for the FHAB, are also applicable to the IUAT, SFISF, CPF, NIAS, CEB. On this basis the NRC staff finds that the preliminary SFISF is consistent with PDC 6.

Fuel Storage and Handling and Radioactivity Control (PDC 61)

In PSAR section 7.3.1.3.4, the applicant discussed certain functions and design criteria applicable to the reinforced concrete structure of the SFISF. PDC 61 relates to: (1) shielding for radiation protection, (2) containment, confinement, and/or filtering systems, (3) residual heat removal, and (4) preventing significant reduction in fuel storage cooling during LBEs. PSAR section 7.3.1.3.4 states that the reinforced concrete SFISF structure provides shielding for radiation protection and passive air flow for residual heat removal. Confinement is provided by the design of the SR fuel pebbles and design of the storage facility to maintain the fuel temperature well below temperatures that could result in a significant release of radionuclides under postulated conditions that could degrade heat transfer to the environment. The SFISF cooling air flow is provided through two vertical intake ducts along the north and south sides of

the SFISF that draw air in above the maximum flood elevation and release it just above the interior floor of the SFISF. PSAR figure 7.3.1-9 shows that the intake ducts run the length of the building (~45 meters) and are 1.5 m wide. The air flow within the SFISF flows from the north and south intakes under and around the stored canisters and exits through a broad central chimney opening in the SFISF roof. The central chimney also runs the length of the building and, as shown in PSAR figure 7.3.1-10, the chimney is 1.5 m wide and includes features to prevent radiation streaming to maintain shielding. PSAR section 3.10.4.1.2 describes an analysis of the SFISF under various flow conditions, including a 100 percent flow blockage, which finds that the building design provided adequate cooling and ample time to detect and correct blockage conditions with maximum fuel storage inventory. The NRC staff's evaluation of this analysis is in section 3.10.1.4 of this SE. Based on the above, the NRC staff concludes that the SFISF provides reliable residual heat removal because the passive cooling air flow through broad openings is resistant to blockage and an analysis demonstrates ample time to recover from a postulated blockage. Therefore, the NRC staff finds that the preliminary NSRST SFISF design is consistent with PDC 61 for the storage of fuel-filled canisters.

Safety Functions

Per PSAR section 7.3.1.3.1, the SFISF is classified as NSRST for preventing excessive challenges to the fuel in response to external hazards due to building collapse.

As discussed under PDCs 1, 2, and 4 above, the design approach for the SFISF implements commercial standards with enhancements applied as special treatments. These enhancements include the use of the SC-IIA design-basis earthquake and supplementing the commercial codes and standards with provisions from nuclear codes and standards. Based on the above, the NRC staff finds that LME's design bases for the SFISF will achieve adequate structural integrity to support performance of its PSF.

Spent Fuel Canisters and Storage Racks

As described in PSAR section 7.3.1.3.4, the purpose of the SFISF is to store the canisters (i.e., the SFCs) and used fuel canisters (UFCs)), which are part of the spent fuel storage system (SFSS) described in PSAR section 1.1.4.3.2. The cylindrical steel canisters are classified as NSRST because they support fuel handling, transfer, and storage functions, including providing supplemental safety functions for storage within the SFISF. The SFISF contains an NSRST steel SFSR system for the safe storage of canisters.

Safety Significance

The canisters and storage rack system maintain fuel pebbles in specific geometries during handling, transfer, and storage. The criticality prevention evaluation in PSAR section 3.9 includes evaluations of filled SFCs and UFCs in a storage array within the SFISF defined by the SFSR positions (PSAR section 3.9.3) and evaluations of a group of four UFCs in interim storage within the CPF (PSAR section 3.9.4). Similarly, a preliminary thermal evaluation of filled SFCs and UFCs in PSAR section 3.10.4 relies, in part, on the arrangement of stored fuel within canisters distributed in the SFSR storage positions to demonstrate acceptable heat removal from each canister with degraded SFISF cooling ventilation flow. These functions all rely on geometry, so the design to retain the geometric arrangement under challenging conditions is important.

Compliance with Applicable PDCs

PSAR section 7.3.1.3.4 states that PDC 1 through 6 are applied to the design of safety-significant SSCs, including the canisters and SFSRs. In addition, PDC 61, 62, and 63 apply to fuel storage and handling areas.

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR section 7.3.1.3.4, which describes that the canisters and SFSRs are designed using industry consensus codes identified in PSAR tables 7.3.1-5 and 7.3.1-6 for the canisters and SFSRs, respectively. The canisters are designed to maintain their integrity in accordance with Nonmandatory Appendices EE & FF of American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code (BPVC) section III, division 3, which allow for plastic deformation within certain strain limits while preventing catastrophic failure during dynamic loading events (e.g., drop from transportation cart). PSAR table 7.3.1-6 indicates that the SFSRs are designed to commercial steel and concrete structural standards supplemented by ASME BPVC section III, division 1, subsection NF, which is a nuclear standard for design of structural supports. PSAR section 7.3.1.3.4 also states that no preliminary enhancements to the code requirements beyond those listed have been identified, although any enhancements applied beyond the baseline will be identified in the OL application.

The NRC staff also reviewed the PSAR for QA treatment. The canisters and SFSRs are governed by the QAP reviewed in SE section 8.1, and QA special treatments will be provided for the OL application. The NRC staff's evaluation of the QAP is in section 8.1 of this SE. Based on the information above, the NRC staff finds that the preliminary design of the canisters and SFSRs are consistent with PDC 1.

Design Basis for Protection Against Natural Phenomena (PDC 2)

PSAR sections 7.3.1.4, 7.3.1.2, and 7.3.1.3 describe that the SFCs and UFCs move between the CPF where canisters are filled and unloaded, the IUAT that provides the transfer path, and the SFISF where canisters are stored, respectively. The SFSRs are located within the SFISF. The canisters and SFSRs are protected against natural phenomena by their location within the NSRST CPF, the NSRST IUAT, and the NSRST SFISF because these NSRST structures are designed to prevent challenges to the SR SSCs (i.e., the fuel pebbles) that they house. As described in PSAR section 3.7, these structures are designed to withstand loads resulting from natural phenomena, including seismic events, hurricanes and tornadoes and associated missiles, external flooding, and extreme climate conditions. The fuel contained within the canisters is protected from the effects of external flooding by the CPF, IUAT, and SFISF design incorporating measures to prevent water intrusion, as described in PSAR section 7.3.1.

The NSRST canisters and SFSR designs include provisions to withstand seismic events in a manner that preserves the dynamic characteristics of the SSCs and achieves performance levels corresponding with their designation as SC- IIA, consistent with PSAR table 3.7-2 and the description in PSAR section 3.7. The NRC staff's evaluation of seismic design bases is presented in section 3.7 of the SE.

Based on the information provided above, the NRC staff finds that the preliminary design of the canisters and SFSRs is consistent with PDC 2.

Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.3.1.3.4 states that the canisters are designed and constructed primarily of metallic materials that are inherently noncombustible and non-fire sustaining. The standards applied to the SFSRs also indicate that they are designed and constructed primarily of metallic materials that are inherently noncombustible and non-fire sustaining. Based on the above information, the NRC staff finds that the design of the canisters and SFSRs is consistent with PDC 3 with respect to minimizing the probability and effect of fires and explosions.

Environmental and Dynamic Effects Design Bases (PDC 4)

The NRC staff reviewed PSAR section 7.3.1.3.4 in which LME states that PDC 4 requirements for the canisters and SFSRs are satisfied by implementing the design and code assessment, the environmental compatibility assessment, and the material qualification plan, as specified in PSAR section 3.8, for NSRST metallic components. As noted above, the canisters are designed to survive falls from the transportation carts and tip-over events while retaining structural integrity to satisfy strain limits specified in Nonmandatory Appendices EE & FF of ASME BPVC section III, division 3. PSAR section 6.1.1 describes the methods to evaluate dynamic effects from internal and external hazards and describes that the credibility of dynamic hazards will be systematically assessed as the design matures.

Based on the above discussion, the NRC staff finds that the preliminary design of the canisters and SFSRs is consistent with PDC 4. The NRC staff will review the final design approach and supporting analyses to demonstrate compliance with PDC 4 regarding protection against dynamic effects, which will be documented in the FSAR to be reviewed for an OL application.

Sharing of Structures, Systems, and Components (PDC 5)

PSAR section 7.3.1.3 describes that the SFISF provides shared storage of fuel discharged from all reactor modules. However, the shared SFISF, which houses individual canisters supported by the SFSRs, does not interface with any reactor systems. Additionally, the operation of the SFISF is independent of each reactor unit and their supporting systems because the key functions of cooling and control of reactivity are accomplished by passive means and the only interface external to the SFISF is the facility outdoor environment as a heat sink. Based on the above review of the design configuration, the NRC staff finds that the shared SFISF fuel storage does not negatively impact the performance of any FHS safety function and the preliminary canister and SFSR design is consistent with PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR section 7.3.1.3.4, which states that periodic inspections of the passive NSRST SFCs, UFCs, and SFSRs are performed in accordance with the reliability, availability, and maintainability (RAM) program, which is evaluated in SE section 8.10. The design provides access for implementing remote preservice and inservice requirements. The NRC staff reviewed the identified potential monitoring and non-destructive examination (MANDE) requirements, including visual examination and health monitoring through temperature and radiation measurements, and finds them appropriate to provide adequate assurance to maintain the functional capability.

Based on the discussion above, the NRC staff finds that the applicant has proposed reasonable monitoring, testing, and inspection programs for the canisters and SFSRs and that these components are designed to provide access for the remote performance of these activities, consistent with PDC 6.

Fuel Storage and Handling and Radioactivity Control (PDC 61)

The NRC staff reviewed PSAR section 7.3.1.3.4, which describes that the SFSR structure elevates the canisters in designated storage locations to preclude degraded cooling air flow due to internal events, such as spilled or ruptured non-safety-related with no special treatment (NST) storage canisters. The evaluation in section 3.10.1.4 of this SE describes crediting the SR fuel pebbles and fuel canister storage geometry of the storage facility (i.e., the combination of the SFISF, SFSRs, and SFCs and UFCs) to maintain the fuel pebble temperature well below temperatures that could result in a significant release of radionuclides under normal operating conditions and postulated conditions that degrade heat transfer to the environment. Although the canisters are sealed for storage, the confinement provided by the canisters is not credited, so the confinement provided by the canisters is an NST function. Based on the NRC staff's evaluation of degraded SFISF cooling scenarios in SE section 3.10.1, the design features of the SFSRs that prevent local blockage of cooling air flow around canisters in storage, and the design of the canisters and SFSRs to retain storage geometry under challenges from internal events and natural phenomena, the NRC staff finds that the preliminary NSRST SFSR, SFC, and UFC designs providing the evaluated fuel storage geometry within the SFISF is consistent with PDC 61.

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

The NRC staff's evaluation in section 3.9.1 of this SE finds that criticality is prevented by the SFSS fuel storage geometry established by the canisters and SFSRs within the SFISF. The NRC staff reviewed PSAR section 7.3.1.3.4, which describes that the design of the SFSRs prevents the accumulation of pebbles around the canisters from events such as canister tip-overs. Based on the NRC staff's evaluation in SE section 3.9 and the above review, the NRC staff finds that the preliminary NSRST SFSR, SFC, and UFC designs are consistent with PDC 62.

Monitoring Fuel and Waste Storage (PDC 63)

The NRC staff reviewed PSAR section 7.3.1.3.4, which states that visual or remote monitoring of spaces under the SFSRs and SFISF ventilation provides appropriate means to detect conditions that may challenge residual heat removal. Considering the margin under degraded cooling conditions to fuel temperatures that would challenge radionuclide retention and the extended time necessary for significant heat up following postulated loss of external cooling events evaluated in PSAR section 3.10.4, the NRC staff finds that visual or remote monitoring of SFISF cooling is acceptable. PSAR section 1.1.4.3.2 describes that the radiation monitoring system detects excessive radiation levels in the SFSS and initiates alarms to notify operators. Based on the review of SFSS monitoring described above, the NRC staff finds that the preliminary SFSS design is consistent with PDC 63.

Safety Function

PSAR section 7.3.1.3.1 states that the SFISF is classified as NSRST for preventing excessive challenges to the fuel stored within the structure. The applicant's preliminary analysis of non-core related sources in PSAR section 3.10, which includes the analyses of the SFSS, states that the PSAR analyses are risk-informed supplemental analyses. The applicant has not identified PRA supported safety functions. The safety function of the SFSS canisters and the SFISF racks is preventing excessive challenges to the fuel. The NRC staff's evaluations of the SFSS design against PDC 61, PDC 62, and PDC 63 provided above indicate that the preliminary design with NSRST canisters and NSRST SFSRs adequately support the NSRST function of preventing excessive challenges to the fuel by maintaining storage geometry consistent with the criticality prevention analyses described in PSAR section 3.9 and the thermal challenge analysis described in PSAR section 3.10.4.1.2. The NRC staff's evaluations of these events are provided in SE sections 3.9.1 and 3.10.1, respectively.

7.3.1.1.4 Canister Processing Facility

As described in PSAR section 7.3.1.4.1, the CPF is a one-level reinforced concrete structure. PSAR figure 7.3.1-11 shows a plan view of the CPF and associated dimensions and PSAR figures 7.3.1-12 and 7.3.1-13 show elevation views for the east and north elevations of the CPF, respectively. PSAR section 3.7 describes, in part, the analysis and design criteria and materials applicable to the CPF. PSAR section 7.3.1.4.1 also provides additional details related to the materials used in the CPF, consistent with the information in PSAR section 3.7.

Safety Significance

As described in PSAR section 7.3.1.4, the CPF provides a physical structure to support the handling and operation of the SFSS for the life of the plant. The CPF provides charging stations for ROST carts, vacuum equipment for contamination removal, an empty SFC staging and rack area, a radiation control room, and an electrical and mechanical equipment room.

PSAR section 5.1.1 addresses the safety classification criteria and associated bases applicable to the CPF including the implementation of design and analysis special treatments that achieve performance levels consistent with the SSCs being supported or protected by the CPF. Per PSAR section 5.5.2 and table 5.5-21, while not modeled in the PRA at the CP stage, the CPF is one of the SSCs that was classified as NSRST for DID adequacy via the IDP. PSAR table 5.5-21 indicates that the PSF for the CPF is to prevent excessive challenges to the fuel in response to external hazards due to building collapse. Additional details on the NRC staff's review of the safety classification are in section 5.1.1 of this SE.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR table 7.3.1-4, which provides the list of codes and standards applicable to the preliminary structural design of the CPF. The design criteria for the NSRST CPF includes the use of commercial codes and standards with enhancements applied as special treatments that include a conservative seismic demand and the use of provisions from nuclear codes and standards including ASCE 4, ASCE 43, and ACI 349-13, among other codes and standards listed in PSAR table 7.3.1-4.

The NRC staff also reviewed the PSAR for QA treatment applicable to the CPF. PSAR section 7.3.1 indicates the applicability of the LMGS QAPD, described in PSAR section 8.1, to the CPF. The CPF is designed, fabricated, constructed, and tested in accordance with the QAP described in part III of the LMGS QAPD and PSAR section 8.1. Additional special treatments related to QA will be specified at the OL application stage. The application of part III of the LMGS QAPD at the CP stage is sufficient as a preliminary special treatment to meet the QA requirements under PDC 1 for the CPF. The NRC staff's evaluation of the LMGS QAPD is provided in the QAPD TR SE and section 8.1 of this SE.

The NRC staff finds that LME's preliminary design bases for the CPF is consistent with PDC 1 because its design bases includes the use of well-established standards, with conservative determination of demands (e.g., seismic demands), the commercial codes and standards used are supplemented as necessary with provisions from nuclear codes and standards, and the CPF is within the scope of LME's QAP.

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 CPF is categorized as an SC-IIA structure and is designed to the **[[**

]] The adequacy of the SC-IIA DRS, as well as the methodology for combining the effects of a seismic event with other effects and loads is evaluated in section 3.7 of this SE. Further, per PSAR section 7.3.1.4.1, the CPF is designed in accordance with the seismic design basis for NSRST structures detailed in PSAR section 3.7. **[[**

]] and LME's consideration of SSI effects as described in PSAR section 3.7, the NRC staff finds that LME's approach for protecting the CPF is acceptable, is consistent with PDC2, and meets the requirements of 10 CFR Part 50, Appendix S, with respect to the consideration of the effects from earthquakes.

The NRC staff reviewed LME's consideration of tornado and hurricane effects on the CPF. Per PSAR sections 3.7 and 7.3.1.4.1, the CPF is designed to withstand tornado and hurricane forces and missile impacts. Additional details on the consideration of tornado and hurricane effects are addressed in PSAR sections 3.7.4.10 and 3.7.4.11 and PSAR table 7.3.1-4. Per these PSAR sections and table, the tornado and hurricane forces and missile impacts are considered in the design of the CPF in accordance with RG 1.76, Revision 1; RG 1.221; and ACI 349, which are applicable to SR structures. Based on LME's use of RG 1.76, RG 1.221, and ACI 349-13 in its consideration of tornado and hurricane effects on the CPF, the NRC staff finds that LME's approach for protecting the CPF from the tornado and hurricane hazards is acceptable and is consistent with PDC 2.

Regarding the consideration of the 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. Further, per PSAR section 7.3.1.4.1, the structural integrity of the CPF 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 7.3.1-4, the NRC staff finds that LME's

approach for protecting the CPF from the flooding hazard is acceptable and is consistent with PDC 2.

PSAR section 3.7 describes the loads and load combinations applicable to NSRST structures, including the CPF. Per the design criteria in PSAR section 3.7, in addition to considering the effects of natural phenomena, the design of the CPF 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 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 loads and load combinations for LMGS structures is addressed in section 3.7 of this SE.

Based on LME's approach for considering the effects of natural phenomena along with the effects from other loads as described above, the NRC staff finds that LME's preliminary design bases for the CPF to protect against natural phenomena consistent with PDC 2.

Fire Protection (PDC 3)

LME's approach for meeting the requirements of PDC 3 and the NRC staff's evaluation described above for the FHAB, are also applicable to the IUAT, SFISF, CPF, NIAS, CEB. On this basis, the NRC staff finds that the preliminary design bases for the CPF are consistent with PDC 3.

Environmental and Dynamic Effects Design Bases (PDC 4)

The NRC staff reviewed PSAR section 7.3.1 and the other PSAR sections mentioned below that address how environmental conditions are considered in the design of the CPF. PSAR sections 3.7.4 and 3.7.5.2 describe the loads and load combinations used in the design of NSRST concrete structures, including the CPF. These loads include normal, severe environmental, extreme environmental, and abnormal loads, as applicable. PSAR section 7.3.1 indicates that NSRST structures, including the CPF, 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 7.3.1 indicates that NSRST structures, including the CPF, 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. The NRC staff's review determined that these environmental effects as well as the design loads, load combinations, material limit states, and deflection values are in accordance with the applicable design codes and standards as discussed in SE section 3.7 and the evaluation of how the design bases for the CPF is consistent with PDCs 1 and 2.

Based on the NRC staff's evaluation described above, the NRC staff finds that LME's preliminary design bases for the CPF are consistent with PDC 4.

Sharing of Structures, Systems, and Components (PDC 5)

The NRC staff reviewed PSAR sections 1.1.3 and 7.3.1, which cover information, in part, related to the sharing of the CPF. PSAR section 7.3.1.4.2 describes that the CPF provides a physical structure to support the handling and operation of the SFSS using ROST carts. The structure serves as a common structure for the canister processing operations of the SFSS and protects

both the spent fuel and equipment inside the building. Per PSAR section 7.3.1.4.1, the CPF is designed to the NSRST criteria in PSAR section 3.7. The NRC staff's evaluation of the NSRST design criteria applicable to the CPF is addressed in SE section 3.7 and above, under the evaluation of how the CPF is consistent with PDC 1, 2, and 4.

Based on the design criteria for the CPF that is focused on protecting the fuel and SSCs within the CPF, the NRC staff concludes that the sharing of the CPF is adequately considered in the design criteria for the CPF, and therefore the NRC staff finds that LME's preliminary design bases for the CPF are consistent with PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

LME's approach for meeting the requirements of PDC 6 and the NRC staff's evaluation described above for the FHAB, are also applicable to the IUAT, SFISF, CPF, NIAS, CEB. On this basis, the NRC staff finds that the preliminary CPF is consistent with PDC 6.

Safety Functions

Per PSAR section 7.3.1.4.1, the CPF is classified as NSRST for preventing excessive challenges to the fuel in response to external hazards due to building collapse.

As discussed under PDC 1, 2, and 4 above, the design approach for the CPF implements commercial standards with enhancements applied as special treatments. These enhancements include the use of the SC-IIA design-basis earthquake and supplementing the commercial codes and standards with provisions from nuclear codes and standards. Based on the above, the NRC staff finds that LME's design bases for the CPF will achieve adequate structural integrity to support the performance of its PSF.

7.3.1.1.5 Nuclear Island Auxiliary Structure

As described in PSAR sections 7.3.1.5 and 7.3.1.5.1, the NIAS is a two-story steel frame open-walled structure with a mezzanine level equipment platform located above the north side of the building. PSAR section 3.7 describes, in part, the analysis and design criteria and materials applicable to NSRST structures such as the NIAS.

Safety Significance

Per PSAR section 7.3.1.5.1, the NIAS a two-story steel frame open-walled structure that houses or allows for passage through SSCs, including the main steam system, the condensate and feedwater system, the start-up and shutdown system (SSS), the nuclear island process water system, the nuclear island instrument air system, the nuclear island cooling water system (NICW), the nuclear island electrical system (NIE), nuclear island fire protection system, and the nuclear island liquid drainage system. The NIAS houses NST, NSRST, and SR systems, and provides needed space for maintenance activities, electrical racks, and batteries.

PSAR section 5.1.1 addresses safety classification criteria and the associated bases applicable to the NIAS including the implementation of design and analysis special treatments that achieve performance levels consistent with the SSCs being supported and protected by the NIAS. Per PSAR section 5.5.2 and table 5.5-21, while not modeled in the PRA at the CP stage, the NIAS is one of the SSCs that was classified as NSRST for DID adequacy via the IDP. PSAR

table 5.5-21 and section 7.3.1.5.1 indicate that the PSF for the NIAS is to prevent excessive challenges to SR SSCs that perform RSF 1.3 within the building in response to external hazards due to building collapse. Additional information on safety classification is addressed in section 5.1.1 of this SE.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR table 7.3.1-7, which provides the list of codes and standards applicable to the preliminary structural design of the NIAS. The design criteria for the NSRST NIAS include the use of commercial codes and standards with enhancements applied as special treatments that include a conservative seismic demand and use of provisions from nuclear codes and standards including ASCE 4, ASCE 43, and AISC N690, among other codes and standards listed in PSAR table 7.3.1-7.

The NRC staff also reviewed the PSAR for QA treatment applicable to the NIAS. PSAR section 7.3.1 indicates the applicability of the LMGS QAPD, described in PSAR section 8.1, to the NIAS. The NIAS is designed, fabricated, constructed, and tested in accordance with the QAP described in part III of the LMGS QAPD and PSAR section 8.1. Additional special treatments related to QA will be specified at the OL application stage. The application of part III of the LMGS QAPD at the CP stage is sufficient as a preliminary special treatment to meet the QA requirements under PDC 1 for the NIAS. The NRC staff's evaluation of the LMGS QAPD is provided in the QAPD TR SE and section 8.1 of this SE.

The NRC staff finds that LME's preliminary design bases for the NIAS are consistent with PDC 1 because its design bases includes the use of well-established standards, conservative determination of demands (e.g., seismic demands), the commercial codes and standards used are supplemented as necessary with provisions from nuclear codes and standards, and the NIAS is within the scope of LME's QAP.

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 adequacy of the SC-IIA DRS, as well as the methodology for combining the effects of a seismic event with other effects and loads is evaluated in section 3.7 of this SE. Further, per PSAR section 7.3.1.5.1, the NIAS is designed in accordance with the seismic design basis for NSRST structures detailed in PSAR section 3.7. As an SC-IIA structure, per PSAR section 3.7.6, the NIAS is designed to meet **[[**

]] and LME's consideration of SSI effects as described in PSAR section 3.7, the NRC staff finds that LME's approach for protecting the NIAS is acceptable, is consistent with PDC2, and meets the requirements of 10 CFR Part 50, Appendix S with respect to the consideration of the effects from earthquakes.

The NRC staff reviewed LME's consideration of tornado and hurricane effects on the NIAS. Per PSAR sections 3.7 and 7.3.1.4.1, the design criteria for the NIAS include consideration of

tornado and hurricane forces and missile impacts. Furthermore, the strategies for protecting SR SSCs within the NIAS include the use of localized protective barriers. Additional considerations of tornado and hurricane effects is addressed in PSAR sections 3.7.4.10 and 3.7.4.11 and PSAR table 7.3.1-7. Per these PSAR sections and table, the tornado and hurricane forces and missiles impacts are considered in the design of the NIAS in accordance with RG 1.76, Revision 1; RG 1.221; AISC N690; and ACI 349 which are applicable to SR structures. Based on LME's use of RG 1.76, RG 1.221, AISC N690, and ACI 349 in its consideration of tornado and hurricane effects on the NIAS, the NRC staff finds that LME's approach for protecting the NIAS and SR SSCs within the NIAS from the tornado and hurricane hazards is acceptable and is consistent with PDC 2.

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 the consideration of hurricane, tsunami, and seiche hazards, among others. Further, per PSAR section 7.3.1.5.1, structural integrity of the NIAS under DBHL flooding conditions is ensured by considering flood loads in the design of the structure as described in section 3.7. PSAR section 7.3.1.5.1 also states that the design of the NIAS accounts for the flooding DBHL to protect SR SSCs located within the NIAS by elevating SR SSCs above the flood height where possible or designing the SR SSCs such that their ability to fulfill an RSF is not impacted if it is submerged. The NRC staff finds these strategies acceptable because they either eliminate the flooding hazards for the SR SSCs with the NIAS or implement design requirements that ensure continued function of the SSCs in submerged conditions. 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 7.3.1-7, and the additional flood protection strategies applicable to SR SSCs within the NIAS, the NRC staff finds that LME's approach for protecting the NIAS and the SSCs within the NIAS from the flooding hazard is acceptable and is consistent with PDC 2.

PSAR section 3.7 describes the loads and load combinations applicable to NSRST structures, including the NIAS. Per the design criteria in PSAR section 3.7, in addition to considering the effects of natural phenomena, the design of the NIAS 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 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 loads and load combinations for LMGS structures, is addressed in section 3.7 of this SE.

Based on LME's approach for considering the effects of natural phenomena along with the effects from other loads as described above, the NRC staff finds that LME's preliminary design bases for the NIAS to protect against natural phenomena are consistent with PDC 2.

Fire Protection (PDC 3)

LME's approach for meeting the requirements of PDC 3 and the NRC staff's evaluation described above for the FHAB, are also applicable to the IUAT, SFISF, CPF, NIAS, CEB. On this basis, the NRC staff finds that the preliminary design bases for the NIAS are consistent with PDC 3.

Environmental and Dynamic Effects Design Bases (PDC 4)

The NRC staff reviewed PSAR section 7.3.1 and the other PSAR sections mentioned below that address how environmental conditions are considered in the design of the NIAS. PSAR sections 3.7.4 and 3.7.5.2 describe the loads and load combinations used in the design of NSRST concrete structures, including the NIAS. These loads include normal, severe environmental, extreme environmental, and abnormal loads, including consideration of HELB loads. PSAR section 7.3.1 indicates that NSRST structures, including the NIAS, 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 7.3.1 indicates that NSRST structures, including the NIAS, 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. The NRC staff's review determined that these environmental effects as well as the design loads, load combinations, material limit states, and deflection values are in accordance with the applicable design codes and standards as discussed in SE section 3.7 and the evaluation of how the design bases for the NIAS are consistent with PDCs 1 and 2.

Based on the NRC staff's evaluation described above, the NRC staff finds that LME's preliminary design bases for the NIAS are consistent with PDC 4.

Sharing of Structures, Systems, and Components (PDC 5)

The NRC staff reviewed PSAR sections 1.1.3 and 7.3.1, which cover information, in part, related to the sharing of the NIAS. PSAR section 7.3.1.5.2 describes that the NIAS provides operations and maintenance space for the NICW, steam piping, SSS, instrument air system and related equipment, fuel storage, and mechanical, electrical, instrumentation, and control systems. Moreover, per PSAR section 1.1.3, the NIAS contains co-located main steam and feedwater lines. Consistent with its PSF, the NIAS protects the SSCs within the NIAS and is designed to the NSRST criteria in PSAR section 3.7. The NRC staff's evaluation of the NSRST design criteria applicable to the NIAS is addressed in SE section 3.7 and above, under the evaluation of how the NIAS is consistent with PDC 1, 2, and 4.

Based on the design criteria for the NIAS that is focused on protecting the fuel and SSCs within the NIAS, the NRC staff concludes that the sharing of the NIAS is adequately considered in the design criteria for the NIAS, and therefore the NRC staff finds that LME's preliminary design bases for the NIAS are consistent with PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

LME's approach for meeting the requirements of PDC 6 and the NRC staff's evaluation described above for the FHAB, are also applicable to the IUAT, SFISF, CPF, NIAS, CEB. On this basis, the NRC staff finds that the preliminary NIAS is consistent with PDC 6.

Safety Functions

Per PSAR section 7.3.1.5.1, the NIAS is classified as NSRST for preventing excessive challenges to SR SSCs within the building in response to external hazards due to building collapse.

As discussed under PDCs 1, 2, and 4 above, the design approach for the NIAS implements commercial standards with enhancements applied as special treatments. These enhancements include the use of the SC-IIA design-basis earthquake and supplementing the commercial codes and standards with provisions from nuclear codes and standards. Based on the above, the NRC staff finds that LME's design bases for the NIAS will achieve adequate structural integrity to support the performance of its PSF.

7.3.1.1.6 Controls and Electrical Building

Safety Significance

As described in PSAR sections 7.3.1.6 and 7.3.1.6.1, the CEB is a two-story concrete building that serves as a center to monitor and control operations of the power plant. The CEB houses control, electrical, and heating, ventilation, and air conditioning (HVAC) systems designated as NST and NSRST, none of which are required for safe shutdown of the plant, but are required for normal plant operation and control, DID strategies, and transient/off normal response.

PSAR section 5.1.1 addresses the safety classification criteria and associated design bases applicable to the CEB including the implementation of design and analysis special treatments that achieve performance levels consistent with the SSCs being supported/protected by the CEB. Per PSAR section 5.5.2 and table 5.5-21, while not modeled in the PRA at the CP stage, the CEB is one of the SSCs that were classified as NSRST for DID adequacy via the IDP. PSAR table 5.5-21 and section 7.3.1.6.1 indicate that the PSF for the CEB is to prevent excessive challenges to NSRST SSCs that perform PSF 1.2.2 within the building in response to external hazards due to building collapse. Additional information on safety classification is addressed in section 5.1.1 of this SE.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR table 7.3.1-8, which provides the list of codes and standards applicable to the preliminary structural design of the CEB. The design criteria for the NSRST CEB include the use of commercial codes and standards with enhancements applied as special treatments that include a conservative seismic demand and the use of provisions from nuclear codes and standards, including ASCE 4, ASCE 43, and ACI 349, among other codes and standards listed in PSAR table 7.3.1-8.

The NRC staff also reviewed the PSAR for QA treatment applicable to the CEB. PSAR section 7.3.1 indicates the applicability of the LMGS QAPD, described in PSAR section 8.1, to the CEB. The CEB is designed, fabricated, constructed, and tested in accordance with the QAP described in part III of the LMGS QAPD and PSAR section 8.1. Additional special treatments related to QA 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 QA requirements under PDC 1 for the CEB. The NRC staff's evaluation of the LMGS QAPD is provided in the QAPD TR SE and section 8.1 of this SE.

The NRC staff finds that LME's preliminary design bases for the CEB are consistent with PDC 1 because its design bases includes the use of well-established standards, with conservative determination of demands (e.g., seismic demands), the commercial codes and standards used

are supplemented as necessary with provisions from nuclear codes and standards, and the CEB is within the scope of LME's QAP.

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 CEB is categorized as an SC-IIA structure and is designed to the SC-IIA DRS **[[**

]] The adequacy of the SC-IIA DRS, as well as the methodology for combining the effects of a seismic event with other effects/loads is evaluated in section 3.7 of this SE. Further, per PSAR section 7.3.1.6.1, the CEB is designed in accordance with the seismic design basis for NSRST structures detailed in PSAR section 3.7. As an SC-IIA structure, per PSAR section 3.7.6, **[[**

]] and LME's consideration of SSI effects as described in PSAR section 3.7, the NRC staff finds that LME's approach for protecting the CEB is acceptable, is consistent with PDC 2, and meets the requirements of 10 CFR Part 50, Appendix S, with respect to the consideration of the effects from earthquakes.

The NRC staff reviewed LME's consideration of tornado and hurricane effects on the CEB. Per PSAR sections 3.7 and 7.3.1.6.1, the design criteria for the CEB include consideration of tornado and hurricane forces and missile impacts. Additional consideration of tornado and hurricane effects is addressed in PSAR sections 3.7.4.10 and 3.7.4.11 and table 7.3.1-8. Per these PSAR sections and table, the tornado and hurricane forces and missile impacts are considered in the design of the CEB in accordance with RG 1.76, Revision 1; RG 1.221; and ACI 349, which are applicable to SR structures. Based on LME's use of RG 1.76, RG 1.221, and ACI 349 in its consideration of tornado and hurricane effects on the CEB, the NRC staff finds that LME's approach for protecting the CEB from the tornado and hurricane hazards is acceptable and is consistent with PDC 2.

Regarding the consideration of the 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. Further, per PSAR section 7.3.1.6.1, structural integrity of the CEB 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 7.3.1-8, the NRC staff finds that LME's approach for protecting the CEB from the flooding hazard is acceptable and is consistent with PDC 2.

PSAR section 3.7 describes the loads and load combinations applicable to NSRST structures, including the CEB. Per the design criteria in PSAR section 3.7, in addition to considering the effects of natural phenomena, the design of the CEB 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 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 loads and load combinations for LMGS structures is addressed in section 3.7 of this SE.

Based on LME's approach for considering the effects of natural phenomena along with the effects from other loads as described above, the NRC staff finds that LME's preliminary design bases for the CEB to protect against natural phenomena are consistent with PDC 2.

Fire Protection (PDC 3)

LME's approach for meeting the requirements of PDC 3 and the NRC staff's evaluation described above for the FHAB, are also applicable to the IUAT, SFISF, CPF, NIAS, CEB. On this basis, the NRC staff finds that the preliminary design bases for the CEB are consistent with PDC 3.

Environmental and Dynamic Effects Design Bases (PDC 4)

The NRC staff reviewed PSAR section 7.3.1, and other PSAR sections mentioned below, that address how environmental conditions are considered in the design of the CEB. PSAR sections 3.7.4 and 3.7.5.2 describe the loads and load combinations used in the design of NSRST concrete structures, including the CEB. These loads include normal, severe environmental, extreme environmental, and abnormal loads, as applicable. PSAR section 7.3.1 indicates that NSRST structures, including the CEB, 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 7.3.1 indicates that NSRST structures, including the CEB, 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. The NRC staff's review determined that these environmental effects as well as the design loads, load combinations, material limit states, and deflection values are in accordance with the applicable design codes and standards as discussed in SE section 3.7 and the evaluation of how the design bases for the CEB are consistent with PDCs 1 and 2.

Based on the NRC staff's evaluation described above, the NRC staff finds that LME's preliminary design bases for the CEB are consistent with PDC 4.

Sharing of Structures, Systems, and Components (PDC 5)

The NRC staff reviewed PSAR sections 1.1.3 and 7.3.1, which cover information, in part, related to the sharing of the CEB. PSAR section 7.3.1.6.1 describes that the CEB provides facilities and space for the equipment and instruments that control and monitor the operations of the plant. Moreover, per PSAR sections 7.3.1.6.1 and 1.1.3, the CEB is shared between the modules. The CEB is designed to the NSRST criteria in PSAR section 3.7 to protect the equipment within the CEB. The NRC staff's evaluation of the NSRST design criteria applicable to the CEB is addressed in SE section 3.7 and above under the evaluation of how the CEB is consistent with PDC 1, 2, and 4.

Based on the design criteria for the CEB that is focused on protecting the equipment within the CEB, the NRC staff concludes that the sharing of the CEB is adequately considered in the design criteria for the CEB, and therefore the NRC staff finds that LME's preliminary design bases for the CEB are consistent with PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

LME's approach for meeting the requirements of PDC 6 and the NRC staff's evaluation described above for the FHAB, are also applicable to the IUAT, SFISF, CPF, NIAS, CEB. On this basis, the NRC staff finds that the preliminary CEB is consistent with PDC 6.

Safety Functions

Per PSAR section 7.3.1.6.1, the CEB is classified as NSRST for preventing excessive challenges to NSRST SSCs within the building in response to external hazards due to building collapse.

As discussed under PDCs 1, 2, and 4 above, the design approach for the CEB implements commercial standards with enhancements applied as special treatments. These enhancements include the use of the SC-IIA design-basis earthquake and supplementing the commercial codes and standards with provisions from nuclear codes and standards. Based on the above, the NRC staff finds that LME's design bases for the CEB will achieve adequate structural integrity to support performance of its PSF.

7.3.1.2 Conclusion

Based on the NRC staff's review of PSAR sections 7.3.1 and 3.7, and other sections of the PSAR as described above, the NRC finds that the information provided in the PSAR gives reasonable assurance that LME's preliminary design of the FHAB, IUAT, SFISF (including the SFCs and storage racks), CPF, NIAS, and CEB meet the applicable regulatory requirements listed under SE section 7.3.1 above. This conclusion is based on:

- LME's identification of the PDCs applicable to the FHAB, IUAT, SFISF (including the SFCs and storage racks), CPF, NIAS, and CEB, including PDCs 1 through 6, PDC 61 applicable to the SFISF, SFCs and storage racks, and PDC 62 and 63 applicable to the SFCs and storage racks.

- The descriptions of design features, design criteria, and special treatments in PSAR sections and 7.3.1 and 3.7 used in the design of these structures and components to meet the applicable PDCs and ensure performance of safety-significant functions.

- The descriptions of materials of construction, general arrangement, and approximate dimensions addressed in PSAR sections and 7.3.1 and 3.7.

- The design basis described in PSAR section 3.7.

7.3.2 Hot Gas Duct System

PSAR section 7.3.2 describes the hot gas duct (HGD) system. The HGD system provides the path for transporting the hot helium from the reactor pressure vessel (RPV) to the shell side of the steam generator pressure vessel (SGPV). The HGD also serves as the boundary between the hot helium exiting the RPV and the cold helium returning to the RPV through the annular region between the HGD outer wall and the cross vessel (CV) inner wall. The annulus between the HGD inner and outer wall is filled with insulation layer that is classified as NST. The HGD

system supports PSF 1.2.2 to maintain core heat removal through active means and PSF 1.2.2.2 to restore active heat removal following the AOO events.

In addition to PDCs 1-6, the following PDC are applicable to the HGD:

- PDC 10 - Reactor design
- PDC-CDC 13 - Instrumentation and control
- PDC-CDC 34 - Residual heat removal

7.3.2.1 Technical Evaluation

Safety Significance

The HGD system is classified as NSRST for DID adequacy as it is relied upon to maintain core heat removal through active means by providing a path for hot helium flow from the RPV to the SGPV. PSAR table 5.5-8 presents a list of LBEs that rely on the HGD and steam generator system (SGS) to perform PSF 1.2.2.2, to restore active heat removal, for assurance of adequate DID. In section 5.5.1 of this SE, the NRC staff determined that the classification of HGD as an NSRST system 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 table 7.3.2-2, which states that HGD metallic SSCs are made of Alloy 800H. PSAR table 7.3.2-3 for non-metallic materials states that the HGD stub is made of graphite. PSAR section 7.3.2.1 states that the HGD system is designed to the 2023 Edition of ASME BPVC section III, division 5, which is endorsed with specific exceptions and limitations by the NRC staff in RG 1.87, Revision 3. Using a nuclear code endorsed by the NRC for NSRST SSCs is a special treatment to ensure quality standards commensurate with the safety significance of the HGD system. In addition, LME performed construction code and material compatibility assessments as described in PSAR section 3.8 and identified three specific areas requiring supplementation to ASME BPVC section III, division 5 provisions for Alloy 800H material: **[[**

]] The NRC staff finds that the selected codes and standards, with the identified supplementation, are consistent with PDC 1 on designing, fabricating, constructing, and testing the safety-significant SSCs to quality standards commensurate with the safety significance of the SSCs. This finding is based on two points. First, ASME BPVC section III, division 5 is endorsed in RG 1.87, Revision 3 as an acceptable approach to ensure structural integrity of the safety-significant SSCs. Second, LME's plan to supplement the code will address the limitations of the code for the material, service life, and service conditions of the preliminary HGD design.

The HGD is designed, fabricated, constructed, and tested in accordance with the QAP described in part III of the LMGS QAPD and PSAR section 8.1. Additional special treatments related to QA 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 QA

requirements under PDC 1 for the NSRST portions of the HGD. The NRC staff's evaluation of the LMGS QAPD is provided in the QAPD TR SE and section 8.1 of this SE.

PSAR table 7.3.2-3 shows that the material for the insulation layer in the annulus between the HGD inner and outer wall is not yet determined and could be ceramic fiber, yttria-stabilized zirconia, calcium silicate, polycrystalline fibrous, or metallic reflective. The PSAR does not discuss what code and standard will be used for the qualification of the insulation layer. During the audit discussion, LME indicated that the insulation layer is classified as NST and is not credited to maintain the temperature of the CV system within the design limit. The CV system interfaces with the HGD system in a concentric arrangement to form a pathway for cold helium flow between the CV inner wall and HGD outer wall. The NRC staff evaluated the CV system in section 7.3.3 of this SE. Since PDC 1 does not apply to NST components, this material selection does not impact the NRC staff's finding for a CP application. The NRC staff notes that thermal analyses for the HGD and CV system will need to be reviewed as part of the OL application to assess whether the analysis methods support the temperature ranges used in the design.

Based on the information above, the NRC staff finds that the preliminary HGD design is consistent with PDC 1.

Design Basis for Protection Against Natural Phenomena (PDC 2)

The NRC staff reviewed PSAR section 7.3.2, which states that the HGD 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 HGD system is categorized as SC-IIA for seismic safety significance to ensure it can perform the associated NSRST PSFs during and following a seismic event. The seismic design requirements associated with SC-IIA 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. PSAR section 7.3.2 also states that the HGD 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. The NRC staff agrees that the HGD components are protected against natural phenomena.

Based on the information above, the NRC staff finds that the preliminary HGD design is consistent with PDC 2.

Fire Protection (PDC 3)

Per PSAR section 7.3.2, the HGD system, except the HGD stub and insulation layer, is constructed primarily of metallic materials that are inherently noncombustible and non-fire sustaining. The HGD stub is made of solid graphite, which cannot sustain combustion or deflagration without a continuous supply of external heat and oxygen (TLR-RES-DE-REB-2026-09, ML26120A199). Moreover, LME states the following: (1) the preliminary analyses indicate a low probability of significant graphite oxidation, and (2) the HGD design effectively eliminates the potential for combustion of the insulation layer, as the insulation is encased within the inner and outer HGD shells that are surrounded by helium.

Based on the above information, the NRC staff finds that the preliminary design of HGD system is consistent with PDC 3.

Environmental and Dynamic Effects Design Bases (PDC 4)

The NRC staff reviewed PSAR section 7.3.2, which states that an environmental compatibility assessment is performed to identify potential material degradation modes using an approach, which is described in PSAR section 3.8 and determined to be acceptable in section 3.8 of this SE. As a result of this assessment, LME identified two testing areas: stress relaxation cracking behavior and compositional changes induced by gas purity variations, (specifically carburization and decarburization phenomena). LME stated that the final special treatments to address these degradation modes are still under development and will be submitted at the OL application stage as the results of the material qualification plan. The NRC staff finds that the preliminary design of the HGD system meets the PDC 4 requirements for being compatible with the environmental conditions associated with different operating conditions because the material qualification plan will address material degradation modes during the service life. PSAR section 7.3.2 states that the DBHL is determined as described in PSAR section 6.1.1 and DBHL related requirements may be applied to the NSRST HGD components to prevent interactions with SR SSCs that would adversely impact the ability of an SR SSC to perform an RSF as described in PSAR section 6.1.3. As evaluated in section 6.1.1 of this SE, LME reasonably characterized the hazards for the CP stage related to PDC 4 requirements and the planned design measures discussed in the PSAR are appropriate to prevent adverse impacts on SR SSCs.

Based on the above discussions, the NRC staff finds that the preliminary design of the HGD system is consistent with PDC 4. Hazards protection strategies, processes, and methods are part of detailed design considerations that will be provided in the FSAR for the NRC staff's review at the OL application stage.

Sharing of Structures, Systems, and Components (PDC 5)

The NRC staff reviewed PSAR section 7.3.2, which states that the HGD SSCs are not shared between reactor modules. HGD components share a foundation system with multiple safety-significant SSCs. The adherence to PDC 5 within the context of shared foundation system is described in PSAR section 3.7.6.3 and the NRC staff's evaluation is described in sections 3.7 and 6.4.1 of this SE. Based on the discussion above, the NRC finds that the preliminary design of the HGD system is consistent with PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR section 7.3.2, which states that periodic inspections of the passive components are performed in accordance with the RAM program (evaluated in section 8.10). The NRC staff reviewed the identified potential MANDE requirements, including temperature and radiation measurements, and helium-side chemistry monitoring and finds them appropriate to provide adequate assurance to maintain the functional capability of the HGD. 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.

The NRC staff will evaluate the RAM program 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 (PSAR section 8.2), which includes passive SSCs that perform functions requiring special treatment for DID adequacy. The NRC staff will review the qualification of the sensors necessary for the HGD to meet the requirements of PDC 6 as well as the adequacy of health monitoring and chemistry control as the only MANDE requirements during the OL application review.

Reactor design (PDC 10): The NRC staff reviewed PSAR section 7.3.2.1, which states that PDC 10 is a higher-level requirement that is met in the HGD system design via application of PDC-CDC 34. The NRC staff finds that this statement is appropriate because it is consistent with the discussion in the Xe-100 PDC TR, which was approved by the NRC staff. As discussed in the basis for PDC 10 in the Xe-100 PDC TR, PDC 10 is a design limit that supports the acceptance criteria for the design and analysis of systems. PDC-CDC 34 describes the requirements for the active heat removal function to meet the specified acceptable system radionuclide release design limit (SARRDL). 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 to meet the requirements for the active heat removal function.

Therefore, the NRC staff finds that the requirements of PDC 10 are satisfied by meeting PDC-CDC 34. The evaluation for compliance of the HGD design to PDC-CDC 34 is performed later in this section.

Instrumentation and Control (PDC-CDC 13)

The NRC staff reviewed PSAR section 7.3.2.1, which states that to support PDC 13, the HGD system has been preliminarily identified as an optimal location for sensors designed to measure hot helium temperature and flow rate to provide indications of the active heat removal function performance. In addition, LME states that both the temperature and mass flow rate of hot helium are monitored parameters in the RPS and investment protection system (IPS) and are controlled variables for the distributed control system (DCS). Based on the above, the NRC staff finds that the preliminary design is consistent with PDC-CDC 13.

Residual Heat Removal (PDC-CDC 34)

The NRC staff reviewed PSAR section 7.3.2.1, which describes the preliminary design for the HGD system and the HGD system connection to the RPV and steam generator (SG) internal assembly flange. The NRC staff finds that the preliminary design of the HGD system provides the pathway for active transport of the hot helium from the RPV to the shell side of the SGPV. In addition, as evaluated earlier in this section, the preliminary design of the HGD system is consistent with PDCs 1, 2, 4 and 6. Meeting PDCS 1, 2, 4, and 6 ensures sufficient structural integrity of the HGD system. Based on the above information, the NRC staff finds that the preliminary design of the HGD system complies with requirements of PDC-CDC 34 by maintaining a path for hot helium to actively transfer fission product decay heat and other residual heat from the reactor core to the SGS. By satisfying PDC-CDC 34, the preliminary design of the HGD system also satisfies PDC 10.

Safety Functions

PSAR section 7.3.2.1 states that the HGD system supports the following NSRST PSFs to demonstrate adequate DID for restoring active heat removal during specific LBEs:

PSF 1.2.2 - maintain core heat removal through active means
PSF 1.2.2.2 - restore active heat removal

LMGS identified PDC-CDC 13 and PDC-CDC 34 as applicable design criteria for the HGD to support PSFs 1.2.2 and 1.2.2.2. In addition, LMGS identified PDCs 1 – 6 and PDC 10 with which the HGD system design complies. The NRC staff finds that these PDC selections are consistent with PSAR section 5.3 and are sufficient to ensure that the design of the HGD system meets the RSFs. The NRC staff finds that the HGD design features are consistent with the PDCs as evaluated in this section.

Based on the evaluation above, the NRC staff finds that the preliminary design of the HGD system provides adequate support for PSFs 1.2.2 and 1.2.2.2, and demonstrates adequate DID for restoring active heat removal during specific LBEs.

PSAR section 7.3.2.3 states that at the preliminary stage of design, no HGD component has been identified as incorporating FOAK technology, design, or configurations that would warrant special treatment testing. The NRC staff will evaluate the continued validity of the FOAK treatment at the OL application when the detailed design information for the HGD is available.

7.3.2.2 Conclusion

The NRC staff concludes that the preliminary HGD is acceptable because:

- Based on the information provided in PSAR section 7.3.2.1, the applicant has met the requirements of 10 CFR 50.34(a)(3)(i) for the HGD by identifying PDCs applicable to the HGD and describing the preliminary design of the HGD.

- Based on the information provided in PSAR section 7.3.2.1 through 7.3.2.3, the applicant has met the requirements of 10 CFR 50.34(a)(3)(ii) by describing how the HGD meets PDCs applicable to the HGD.

- Based on the information provided in PSAR section 7.3.2.1 through 7.3.2.3, the applicant has met the requirements of 10 CFR 50.34(a)(3)(iii) by providing the general materials of construction and component arrangement in the HGD.

7.3.3 Cross Vessel System

PSAR section 7.3.3 describes the CV system. The CV system comprises a horizontally oriented pressure vessel and appurtenances that connect the RPV to the SGPV. The CV system functions as part of the helium pressure boundary (HPB) to maintain primary system pressure, and interfaces with the HGD system in a concentric arrangement to provide a pathway for transport of the cold helium from the shell side of the SGPV to the RPV.

In addition to PDCs 1 - 6, the following PDCs are applicable to the CV:

- 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-CDC 34 - Residual heat removal

7.3.3.1 Technical Evaluation

Safety Significance

The CV system is classified as NSRST for DID adequacy. PSAR table 5.5-16 presents a list of LBEs that rely on the CV to perform PSF 2.7 to maintain HPB pressure integrity for assurance of adequate DID. In section 5.5.1 of this SE, the NRC staff determined that the classification of CV as NSRST system 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 table 7.3.3-2, which shows that the CV shell and flanges are made of SA-508, Grade 3, Class 1 material and the fasteners are made of SA-540, Grade B23, Class 3 material. The PSAR also states that the design temperature is 285°C. PSAR section 7.3.3.1 states that the CV system is designed to the 2023 Edition of ASME BPVC section III, division 5, which is endorsed by the NRC staff in RG 1.87, Revision 3. Using a nuclear code endorsed by the NRC staff for NSRST SSCs is a special treatment to ensure that the quality standards commensurate with the safety significance of the CV system. PSAR section 7.3.3.1 also states that LME performed construction code and material compatibility assessments as described in PSAR section 3.8, and the assessments conclude that neither supplementation to the code requirements nor mitigation to any material degradation mode is necessary, considering that no CV SSCs operate at or above elevated-temperature design conditions as defined in ASME BPVC section III division 5, table HAA-1130-1. The CV is designed, fabricated, constructed, and tested in accordance with the QAP described in part III of the LMGS QAPD and PSAR section 8.1. Application of part III of the LMGS QAPD at the CP stage is sufficient as a preliminary special treatment to meet the QA requirements under PDC 1 for the NSRST portions of the CV. 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 above information, the NRC staff finds that the preliminary design of the CV system is consistent with the requirements of PDC 1.

Design Basis for Protection Against Natural Phenomena (PDC 2)

The NRC staff reviewed PSAR section 7.3.3, which states that the CV system is designed to withstand the effects of earthquakes in accordance with the designated seismic category in PSAR table 3.7-2 and the methods described in PSAR sections 3.7. As stated in PSAR

table 3.7-2, the CV system is categorized as SC-IIA for seismic safety significance to ensure that it can perform the associated NSRST PSFs during and following a seismic event. The seismic design requirements associated with SC-IIA 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. The PSAR also states that the CV components are protected against natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions by their location within the SST. The NRC staff agrees that CV components are protected from natural phenomena as described in section 6.4.1 of this SE. Based on the information above, the NRC staff finds that the CV design is consistent with PDC 2.

Fire Protection (PDC 3)

Per PSAR section 7.3.3, the CV system is constructed of metallic materials that are inherently noncombustible and non-fire sustaining. The CV is located entirely within the concrete SST, with low combustible loading and limited fire hazard potential in the immediate vicinity. In addition, PSAR section 7.3.3 states that the CV system is protected by a comprehensive plant FPP, 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's evaluation of the LME FPP is presented in section 8.6 of this SE. Based on the above information, the NRC staff finds that the preliminary design of the CV system is consistent with PDC 3.

Environmental and Dynamic Effects Design Bases (PDC 4)

The NRC staff reviewed PSAR section 7.3.3, which states that an environmental compatibility assessment is performed using the approach described in PSAR section 3.8, which is determined to be acceptable in section 3.8 of this SE. The assessment identified no potential material degradation modes applicable to the CV system. PSAR section 7.3.3.1 also states that analyses will be performed to demonstrate adequate CV protection from dynamic hazards, including an evaluation of external pressure loads on the CV from rapid compartment pressurization and an assessment of impingement generated by helium discharge. In addition, LME states that design measures are implemented to mitigate dynamic effects, including the incorporation of component supports and shielding as well as strategic equipment location.

Based on the above discussions, the NRC staff finds that the preliminary design of the CV system is consistent with PDC 4. Hazards protection strategies, processes, and methods are part of the detailed design considerations that will incorporate insights from more detailed deterministic and probabilistic hazards assessments in the FSAR. The analysis and design approach to demonstrate that dynamic effects are appropriately addressed will be documented in the FSAR at the OL application stage.

Sharing of Structures, Systems, and Components (PDC 5)

The NRC staff reviewed PSAR section 7.3.3, which states that the CV SSCs are not shared between reactor modules. The CV components share a foundation system with multiple safety-significant SSCs. The adherence to PDC 5 within the context of the shared foundation system is described in PSAR section 3.7.6.3. The NRC staff evaluated this description and finds

it acceptable as described in sections 3.7 and 6.4.1 of this SE. Based on the discussion above, the NRC finds that the preliminary design of the CV system is consistent with PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR section 7.3.3, which states that periodic inspections of the passive components are performed in accordance with the RAM program, which is evaluated 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 health monitoring through temperature and radiation measurements and helium-side chemistry monitoring and finds them appropriate to provide adequate assurance to maintain the functional capability of the CV system. 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.

The NRC staff will evaluate the RAM program 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 (PSAR section 8.2), which includes passive SSCs that perform functions requiring special treatment for DID adequacy

Reactor Design (PDC 10)

The NRC staff reviewed PSAR section 7.3.3, which states that PDC 10 is a higher-level requirement that is met in the CV system design via application of PDC-CDC 34. The NRC staff finds that this statement is appropriate because it is consistent with the NRC-approved Xe-100 PDC TR. As discussed in the basis for PDC 10 in the Xe-100 PDC TR, PDC 10 is a design limit that supports the acceptance criteria in PDC-CDC 34, which describes the function provided by active heat removal in meeting the SARRDL. Therefore, PDC 10 requirements are satisfied by meeting PDC-CDC 34. The evaluation for compliance of the CV design to PDC-CDC 34 is performed later in this section.

Instrumentation and Control (PDC-CDC 13)

The NRC staff reviewed PSAR section 7.3.3, which states that the preliminary design will have sensors located within the SGPV outlet plenum upstream of the CV system inlet in the SG upper shell cavity to measure cold helium temperature. No sensors will be located in the CV system out of practicality considerations. Based on the information provided in the PSAR, the NRC staff finds that the preliminary design is consistent with PDC 13 because the hot and cold helium temperature, primary system pressure and humidity, and helium mass flow rate are sufficiently monitored with sensors located throughout other portions of the HPB primary helium flow path that are interconnected with the CV helium flow path.

Reactor Helium Pressure Boundary (PDC 14)

As discussed earlier in this section, the preliminary design of the CV system complies with PDCs 1, 2, 4, and 6 to ensure structural integrity of the CV system. In addition, per PSAR section 7.3.3.1, the CV shell and flanges are fabricated as a single forged component, which precludes pressure boundary welds. Based on this information, the NRC staff finds that the

preliminary design of the CV system ensures an extremely low probability of abnormal leakage and unacceptable moisture ingress and thus is consistent with PDC 14.

Reactor Helium Pressure Boundary Design (PDC-CDC 15)

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

Functional Containment Design (PDC-CDC 16)

As discussed earlier in this section, the preliminary design of the CV system complies with PDCs 1, 2, 4, and 6 to ensure structural integrity of the CV system. The NRC staff finds that these preliminary design features of the CV system are sufficient to provide a barrier as part of the reactor functional containment to control the release of radioactivity to the environment and thus are consistent with PDC-CDC 16.

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

The NRC staff reviewed PSAR section 7.3.3, which states that the HPB leakage detection is primarily achieved with the design of appropriate interfaces with plant sensors and associated instrumentation and control (I&C) systems that monitor key process parameters and facilitate detection and isolation of helium leakage from the HPB. The leakage detection requirements of PDC-CDC 30 are met largely via the appropriate application of PDC-CDC 13 as a design requirement for those I&C systems. Based on this information, the NRC staff finds that the preliminary design of the CV system is consistent with PDC-CDC 30.

Fracture Prevention of Reactor Helium Pressure Boundary (PDC 31)

As evaluated in section 3.8 of this SE, the NRC staff finds that the structural integrity evaluation method and material qualification plan in 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, design-basis events (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 NRC staff finds that 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 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 CV system is consistent with PDC 31 for the CP application.

Residual Heat Removal (PDC-CDC 34)

The NRC staff reviewed PSAR section 7.3.3.1, which describes the preliminary design for the CV system. The annular region formed by the HGD outer wall and the CV shell inner wall provides a flow path for the cold helium returning from the SGPV to the RPV, thus facilitating the active heat transfer from the reactor core to the SGS. Based on this information, the NRC staff finds that the preliminary design of the CV system satisfies PDC-CDC 34 by providing a helium

flow path to facilitate the active heat transfer of the fission product decay heat and other residual heat from the reactor core to the ultimate heat sink to meet the SARRDL. 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 to meet the requirements for the active heat removal function. In addition, as evaluated earlier in this section, the preliminary design of the CV system is consistent with PDCs 1, 2, 4 and 6. This ensures sufficient structural integrity of the CV system.

Based on the above information, the NRC staff finds that the preliminary design of the CV system complies with requirements of PDC-CDC 34, by maintaining a path for cold helium flow to facilitate effective active heat removal from the core and satisfy the SARRDL. By satisfying PDC-CDC 34, the preliminary design of the CV system also satisfies the requirements of PDC 10.

Safety Functions

PSAR section 7.3.3.1 states that the CV system supports the following NSRST PSFS to demonstrate adequate DID for the indicated HPB radionuclide retention and pressure integrity functions during specific LBEs:

PSF 1.2.2 – maintain core heat removal through active means
PSF 1.2.2.2 – restore active heat removal
PSF 2 – retain radionuclides in the HPB
PSF 2.7 – maintain the HPB pressure integrity
PSF 2.7.1 – control the frequency of small depressurization events
PSF 2.7.2 – control the frequency of medium depressurization events
PSF 2.7.3 – control and frequency of large depressurization events

LMGS identified PDC-CDC 15, PDC-CDC 16, PDC-CDC 30, and PDC-CDC 34 as applicable design criteria for the CV to support PSFs 2 and 2.7. In addition, LME identified that the CV system must comply with PDCs 1 – 6, 10, 13, 14, and 31. The NRC staff finds that these PDC selections align with PSAR section 5.3, and are sufficient to ensure that the design of the CV system meets the NSRST PSFs. Based on the evaluation above, the NRC staff finds that the preliminary design of the CV system provides adequate support for the NSRST PSFs identified and demonstrates adequate DID for the indicated HPB radionuclide retention and pressure integrity functions during specific LBEs.

7.3.3.2 Conclusion

The NRC staff concludes that the preliminary CV is acceptable because:

- the applicant has met the requirements of 10 CFR 50.34(a)(3)(i) for the CV by identifying the PDC applicable for the CV and describing the preliminary design of the CV in PSAR section 7.3.3.1;

- the applicant has met the requirements of 10 CFR 50.34(a)(3)(ii) by describing how the CV meets the PDC applicable for the CV in PSAR section 7.3.3.1 through 7.3.3.3; and

- the applicant has met the requirements of 10 CFR 50.34(a)(3)(iii) by providing the general materials of construction and component arrangement in the CV in PSAR section 7.3.3.1 through 7.3.3.3.

7.3.4 Start-up and Shutdown System

PSAR section 7.3.4 describes the SSS. The SSS provides feedwater supply and decay heat removal during startup and shutdown and during certain LBEs. The SSS manages SG outlet pressure, produces steam to warm the conventional island, maintains steam flow to the condenser system (CDS), and transitions SGs to steaming during reactor startup. The SSS also sustains steam delivery to the CDS for heat rejection during shutdown. The SSS remains in standby mode during normal operation, available to supply feedwater when the main feedwater system is unavailable or to remove decay heat if the CDS is out of service.

The SSS is designed to provide an active means of core heat removal, remove water and steam from the SG during certain LBEs that involve water/steam ingress, and ensure that the SARRDLs are not exceeded during AOOs. The system is shared between four plant modules. The SSS can remove the decay heat load for all four units by cycling between the four SGs in roughly 1-hour intervals for first three days and supports all four SGs simultaneously after about 3 days.

During normal power operations, the SSS is isolated from the SGS and operates in a standby mode by continuously circulating water in the system to ensure that the SSS feedwater temperature, pressure, and chemistry are sufficient for potential injections into the SGS. The primary components of the SSS are:

- Blowdown tanks: two tanks, each 100 percent redundant,
- SSS feed pumps: two pumps, each 100 percent redundant,
- SSS feed heat exchangers: two 50 percent capacity air-cooled feed heat exchangers,
- Attemperation tank: one tank to provide net suction head for attemperation pumps,
- Attemperation pumps: two pumps, each 100 percent redundant,
- Attemperation heat exchangers: two 50 percent capacity air-cooled attemperation heat exchangers,
- SGS to SSS steam header: two steam headers, one to direct steam from any SG tube A-bundles and another to direct steam from any SG tube B-bundles,
- SGS to SSS water header: two water headers, one to direct water from any SG tube A-bundles and another to direct water from any SG tube B-bundles,
- SSS feed water header: one feed water header to direct feed flow from SSS feed pumps to SG tube bundles.

The SSS system interfaces with the following SSCs:

- SGS: to provide feedwater and accept steam from the SGS,
- Helium circulator system (HCS): to allow for active decay heat removal during certain LBEs,
- DCS: to provide controls for the SSS during normal operation and during certain LBEs,
- CDS: for condensing during normal startup and operations and during certain LBEs when available,
- Condensate system (CDCN): to make up feedwater inventory via attemperation tanks

The industry standard codes and materials used to construct the SSS system are provided in PSAR table 7.3.4-1 and the process flow chart for the SSS is provided in PSAR figure 7.3.4-1.

The system-specific guidance for the evaluation of SSS is as follows:

- DANU-ISG-2023-01, “Material Compatibility for non-Light Water Reactors,” (ML22203A175);
- RG 1.87 Revision 3, “Acceptability of ASME Code Section III, Division 5, ‘High Temperature Reactors’” (ML25176A084); and
- RG 1.29, Revision 6, “Seismic Design Classification for Nuclear Power Plants.” (ML21155A003).

In addition to PDC 1-6, the following system-specific PDC are applicable to the SSS:

- PDC 10 - Reactor design
- PDC-CDC 13 - Instrumentation and control
- PDC-CDC 34 - Residual heat removal.

7.3.4.1 Technical Evaluation

Safety Significance

The SSS along with the associated subsystems is classified as NSRST. The SSS system is relied upon to demonstrate adequate DID to restore active heat removal and remove moisture following isolation of moisture sources during AOOs, DBEs, and beyond-design-basis events (BDBEs), as listed in PSAR table 5.5-9. The SSS is designed to:

- provide an active means of fission product decay heat transfer and other residual heat transfer from the reactor core at a rate that the SARRDLs are not exceeded during AOOs, and

- remove moisture sources from the SG at a rate that precludes lifting of safety relief valves during certain LBEs that involve moisture ingress and ensure that SARRDLs are not exceeded.

The failure of the SSS leads to loss of active heat removal via forced cooling, and the reactor transitions from forced convection to a passive heat-transfer regime to transfer heat from the core and the surrounding structures to the RPV. The heat transfer from the RPV to the atmosphere is provided by the passive and active reactor cavity cooling system (RCCS).

The PSAR presents the NSRST PSFs supported by the SSS to demonstrate adequate DID. As confirmed by the NRC staff via RCI (ML26239A157), LME is in the process of incorporating the results of updated risk-significance evaluations following the NEI 18-04 methodology. The preliminary results of these evaluations indicate that the SSS is risk significant, which would mean that the failure of the prevention or mitigation function of the SSS could lead some of the LBEs crediting the SSS to exceed the F-C target. In section 5.1 of this SE, the NRC staff finds that confirmation of final risk significance can reasonably be left to the OL stage in accordance with 10 CFR 50.35(a)(2).

The PSAR identifies the NSRST portions of the SSS with the overall system being categorized as NSRST. The PSAR did not identify any NST components in the SSS system.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

For the SSS, all components are identified as NSRST and will be designed to the industry consensus codes and standards listed in PSAR table 7.3.4-1, e.g., ASME B31.1, and discussed in PSAR section 3.8. As stated in PSAR table 7.3.4-1, some of the materials, codes of construction and design conditions have not yet been determined for the SSS components. The NRC staff finds the level of detail for the design conditions, materials, construction codes, and special treatments to be appropriate for the preliminary design of the SSS for the CP application. The NRC staff will review detailed design and construction specifications for the final design of the SSS, at the OL application stage.

The PSAR identifies that the SGS to SSS steam header and associated valves operate at low temperature under normal conditions and will be designed to ASME B31.1. However, during startup, shutdown, and some DBEs, the temperatures exceed the conditions defined in ASME III.5, table HAA-1301. To address these high-temperature excursions, the PSAR states that special treatments may include the evaluation of creep-fatigue effects on component cyclic life and demonstration that the cumulative elevated-temperature service hours bound the time basis for ASME B31.1 allowable stress values or an adjustment to the allowable stresses accordingly.

The PSAR states that the safety-significant NSRST SSS SSCs are designed, fabricated, constructed, and tested to quality standards commensurate with their NSRST function and are governed by the QAP described in PSAR section 8.1. The QAP is evaluated in section 8.1 of this SE.

Special treatments for SSS components in general have not yet been identified, though the PSAR states that preliminary special treatments under consideration include 100 percent

volumetric examination (UT or radiographic testing (RT)), pressure boundary fusion welds, and full penetration welds without permanent backing strips. The structural evaluation of the SSS metallic SSCs is provided in PSAR section 3.8, and the acceptability of the evaluation is discussed in section 3.8.1 of this SE. The PSAR states that the metallic components listed in PSAR table 7.3.4-1 operate at low-temperature design conditions (under normal, standby operation) and their structural failure modes are adequately addressed by the use of ASME section VIII, ASME B31.1 and API 2000.

Based on its review of the information provided, the NRC staff finds that the preliminary design of the SSS is consistent with PDC 1.

Design Basis for Protection Against Natural Phenomena (PDC 2)

The SSS pumps, blowdown tanks, and one attemperation tank are located within the NIAS building while the four air-cooled heat exchangers are located adjacent to the NIAS building. The PSAR indicates that the NIAS and adjacent support structures will be designed to withstand the effects of natural phenomena, consistent with the DBHL requirements for NSRST components identified in PSAR sections 6.1.1 and 6.1.3. The NSRST SSS SSCs are designed to withstand seismic events and achieve performance levels corresponding to their designated category, as shown in PSAR table 3.7-2 and described in PSAR section 3.7. The NRC staff's evaluation of the seismic design basis is presented in section 3.7 of this SE. Further, PSAR section 7.3.1.5.1 states that the structural integrity of the NIAS under DBHL flooding conditions is ensured by considering flood loads in the design of the structure as described in PSAR section 3.7. The PSAR states that the four air-cooled heat exchangers located outside of the NIAS building will be designed to ensure that missiles or external flooding generated by applicable natural phenomena do not compromise their ability to perform any PSF. The PSAR further states that hazard protection strategies, processes, and methods are part of detailed design considerations that will incorporate insights from more detailed deterministic and probabilistic hazards assessments in the FSAR. The NRC staff finds this to be acceptable given that the preliminary information presented above shows reasonable assurance of protection against natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunamis, and seiches without loss of capability of the SSS to perform its safety-significant functions.

Based on the on the information above, the NRC staff finds that the SSS is consistent with PDC 2.

Fire Protection (PDC 3)

The SSS components will be constructed using noncombustible or fire-resistant materials to the extent practicable. The SSS components are located within or adjacent to the NIAS in areas where fixed combustible materials are negligible and transient combustibles are administratively controlled. PSAR section 8.6 provides a description of the FPP including a list of program elements that are implemented in an integrated manner to meet the requirements of 10 CFR 50.48(a) as well as PDC 3 to minimize the probability and effects of fires and explosions. These elements include, in part, building and system design and layout, including the appropriate use of fire-resistant assemblies, such as fire barriers, fire doors, fire dampers, and fire penetration seals. PSAR section 8.6 also indicates that the FPP elements will be described in the operational FPP to be developed in conjunction with and described in the application for an OL. Moreover, the PSAR states that the FPP will be designed in full

conformance with the portions of RG 1.189 that are technically relevant to the design. The acceptability of the FPP is described in section 8.6 of this SE.

Based on the review of the preliminary design features of the SSS and the programmatic commitments described in PSAR section 8.6, the NRC staff finds that the fire protection measures for the SSS are consistent with the intent 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 7.3.4.1, which states that an environmental compatibility assessment is performed using the structural evaluation methodology described in PSAR section 3.8. The acceptability of the methodology is described in section 3.8 of this SE. The assessment identified no potential material degradation modes applicable to the SSS system. PSAR section 7.3.4.1 states that analyses will be performed to demonstrate adequate SSS protection from dynamic hazards, including an evaluation of external pressure loads on the SSS from rapid compartment pressurization and an assessment of impingement generated by helium discharge. In addition, design measures are implemented to mitigate dynamic effects, including the incorporation of component supports and shielding as well as strategic equipment location.

The NRC staff reviewed the PSAR statement that hazard protection strategies, processes, and methods are part of detailed design considerations which will incorporate insights from more detailed deterministic and probabilistic hazards assessments in the FSAR, and that the analysis and design approach to demonstrate that dynamic effects are appropriately addressed will be presented at the OL application stage. The NRC staff finds LME's approach to ensure adequate protection of the SSS from dynamic effects to be acceptable because it provides a structured, commitment-backed process to evaluate dynamic hazards as the design matures and ensures that the final FSAR will contain the analyses necessary to demonstrate compliance with PDC 4.

Based on the review of the information provided, the NRC staff concludes that there is sufficient detail in the PSAR to demonstrate that the SSS is designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, AOOs, DBEs, and DBAs commensurate with the safety significance of the functions to be performed and is consistent with PDC 4.

Sharing of Structures, Systems, and Components (PDC 5)

The NRC staff reviewed PSAR section 7.3.4, which states that the SSS design includes common components that provide cooling and blowdown capability to each plant module and thus the SSS is shared by all four plant modules.

During certain LBEs, with CDS unavailable and a single SG isolated, the SSS feed pump supplies water directly to the isolated SG, and its steam is routed to the blowdown tanks (BDTs), where it is cooled and pressure-controlled by the SSS feed heat exchangers. The pressure and temperature in each BDT are maintained via the attemperation heat exchangers, tank, and pumps, which maintain active core heat removal. For AOOs with CDS available, SG steam can be routed through the BDTs to the CDS to avoid affecting the common steam header, SG units, and other operating modules. During a complete loss of feedwater AOO affecting all four SGs, the SSS operates similar to the single SG isolation scenario without conventional island support. Due to the limited initial heat removal capacity and the large

thermal inertia, the SSS sequentially injects feedwater into one SG at a time, cycling between the four SGs in roughly 1-hour intervals. The helium circulators in each of the four units align to the sequential cycling of the SSS. The SSS can remove the decay heat load for all four units by cycling between them and is able to support all four SGs simultaneously after about 3 days. Similarly, during a SG tube leak or SG tube rupture event with successful SG isolation, the steam mixture in the isolated SG is directed to the BDTs. Because the steam mixture is superheated, one attemperation pump is needed to pump feed water from the SSS to the BDTs to quench the steam and maintain pressure.

The NRC staff compared the dose consequences for the LBEs ending in SSS cooling for a single module with the dose consequences for the LBEs ending in SSS cooling for all four modules to assess the impact of sharing through cycling between the four SGs to ensure that there is margin to the F-C target. As discussed in SE section 3.2, the applicant did not perform consequence modeling for all event sequences identified in the PSAR and chose analyses for other LBEs as representative or bounding of similar events. While the NRC staff determined that this approach is acceptable at the CP stage when design information is limited, additional analyses are expected at the OL application stage to ensure that the impacts of sharing the SSS between four modules through sequential cycling is appropriately accounted for in the consequence analysis results and does not impact the ability of the SSS to perform its NSRST PSFs. Based on the review of this information, the NRC staff finds that the preliminary evaluation of the SSS system shows that sharing between modules does not inhibit its ability to perform its safety functions, commensurate with the safety significance of the functions to be performed. Therefore, the NRC staff finds that the design of the SSS is consistent with PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR section 7.3.4, which states that periodic inspections of the passive components are performed in accordance with the RAM program, which is evaluated in SE section 8.10. The SSS design provides access for implementing remote preservice and inservice requirements. The NRC staff reviewed the identified potential MANDE requirements, including visual examination and secondary-water-side chemistry monitoring. Because the design provides access and proposed monitoring and inspection addresses credible degradation mechanisms, the NRC staff finds that the preliminary inspection and monitoring measures provide reasonable assurance that the functional capability of the SSS will be maintained.

LME states that functional testing of the active components will be performed in accordance with the maintenance program that is evaluated in SE section 8.3. However, as discussed in the safety significance subsection above, LME is in the process of incorporating the results of updated risk-significance evaluations following the NEI 18-04 methodology. The preliminary results of these evaluations indicate that the SSS is risk significant. If the active functions of the SSS are risk significant, based on the scope of the IST program described in PSAR section 8.4, the risk-significant SSCs would be expected to be included in the IST program. For non-risk-significant active SSCs, the NRC staff notes that the maintenance program does not specify a method for developing and implementing an IST program, such as ASME OM-2. The NRC staff will evaluate the operational program scoping of SSCs during the OL application based on final risk-significance determinations. The NRC staff reviewed potential IST requirements, including exercise testing and diagnostic testing. The planned functional testing is expected to encompass the conditions and ranges associated with operation of the SSS SSCs. The adequacy of the test plan, including test conditions, measurements, and monitoring, will be

evaluated during the OL review. In addition, LME states that the preoperational testing, as described in PSAR section 12.1.1 and evaluated in SE section 12.1, will be performed to provide a baseline for the inservice inspections.

Based on the discussion above, the NRC staff finds that the risk-significant active SSS SSCs are consistent with 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 DID adequacy.

Reactor Design (PDC 10)

The NRC staff reviewed PSAR section 7.3.4, which states that PDC 10 is a higher-level requirement that is met for the SSS via application of PDC-CDC 34 and system-level CDCs aligned with NSRST PSF 1.2.2, maintain core heat removal through active means; PSF 1.2.2.2, restore active heat removal; and PSF 1.3.2, remove water/steam following isolation of water/steam sources. The SSS is designed with appropriate margin to ensure it performs the required PSFs through:

- appropriately designated quality, seismic, and ASME code classifications,
- application of approved codes and standards commensurate with NSRST PSFs applied to SSS, and
- selection of materials with consideration for operating conditions that the SSS components would experience during normal operations and certain LBEs.

The NRC staff reviewed the details of SSS operation and performance provided in PSAR section 7.3.4.2 and finds them to be acceptable. The acceptability of the three elements used to meet the PSFs applied to the SSS are discussed in the PDC-CDC 34 description below.

Based on the review of the information provided, the NRC staff concludes that PDC 10 requirements for SSS are satisfied by meeting PDC-CDC 34 and the system-level CDCs align with NSRST PSFs 1.2.2, 1.2.2.2 and 1.3.2.

Instrumentation and Control (PDC-CDC 13)

The NRC staff reviewed PSAR section 7.3.4, which states that the I&C provided by the DCS and IPS systems facilitate the performance of NSRST PSFs for the SSS during normal operation and certain LBEs. The instrumentation is provided to monitor variables over their anticipated ranges during normal operation and certain LBEs and maintain those variables within the prescribed ranges.

Based on the review of the description of the I&C provided for the SSS and their 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 SSS is consistent with PDC-CDC 13.

Residual Heat Removal (PDC-CDC 34)

As discussed for PDC 10 above, the NRC staff reviewed the SSS to ensure that it is designed with appropriate margin to perform the required PSFs through:

- appropriately designated quality, seismic, and ASME code classifications,
- application of approved codes and standards commensurate with the NSRST PSFs applied to the SSS, and
- selection of materials with consideration for operating conditions that the SSS components would experience during normal operations and certain LBEs.

The SSS is classified as seismic category II (SC-II) and subject to graded QA controls in accordance with the approved QAP commensurate with the role of the SSS in performing DID. PSAR table 7.3.4-1 identifies the design conditions, materials, and the applicable codes and standards for the NSRST SSS components.

The NRC staff reviewed the PSAR statement that the SGS to SSS steam header and associated valves operate at low temperature under normal conditions. However, during startup, shutdown, and some DBEs, the temperatures exceed the low-temperature limits defined in ASME BPVC III.5, table HAA-1301-1. The codes and standards, e.g., B31.1, and special treatments associated with these components are discussed under the NRC staff's evaluation of PDC 1 for the SSS.

The PSAR identifies a dissimilar metal joint between the SSS internals and the SSS feedwater header. The joint in question is between austenitic SS and ferritic steel and states that this joint will operate at low-temperature service conditions as defined in ASME BPVC III.5, table HAA-1130-1. The NRC staff reviewed PSAR table 7.3.4-1, which identifies the construction code for the SSS internals and the SSS feedwater header as B31.1 and states that the welds will be code compliant. Based on its review of the description of the codes and standards provided for the SSS and their application commensurate with the NSRST PSFs applied to the SSS along with the selection of the materials for the operating conditions for the SSS, the NRC staff finds that the preliminary design of the SSS is consistent with PDC-CDC 34.

Safety Functions

<u>Heat Removal Function</u>

PSAR section 7.3.4.1 states that the SSS is designed to meet the following PSF through thermal transportation capabilities during normal operations and certain LBEs:

- PSF 1.2.2 - maintain core heat removal through active means
  - PSF 1.2.2.2 - restore active heat removal

The SSS system applies the plant-level PDC-CDC 34 to support performance of NSRST PSFs 1.2.2 and 1.3.2. The following system-level CDCs are applied to the SSS to provide heat removal during AOOs and to ensure that SARRDLs are not exceeded:

- CDC for PSF 1.2.2.2.1
- CDC for PSF 1.2.2.2.2
- CDC for PSF 1.2.2.2.3
- CDC for PSF 1.2.2.2.4

The SSS uses redundant blowdown tanks to receive steam from the SGs for decay heat removal and steam containment during certain LBEs, along with redundant feed pumps to supply feedwater when the main feedwater system is unavailable. The feed heat exchangers cool the flow from the blowdown tanks before injection into the SGs. An attemperation system, consisting of a tank, two fully redundant pumps, and two half capacity air-cooled heat exchangers, provides capability to reduce the inlet temperature, maintain blowdown tank pressure, and prevent boil-off by keeping water temperature below 100°C.

The SSS system operates the DCS automatically during normal operations and certain LBEs using the controls, logic circuits, and interlocks provided by the DCS, requiring no operator action or displays to support NSRST PSFs 1.2.2.2.1 through 1.2.2.2.3. Operator actions and displays are needed for the SSS to support the NSRST PSF 1.2.2.2.4 to restore active heat removal during certain LBEs that involve water/steam ingress. The completion of automatic actuation signals processed by the IPS allow for operator actions, via the DCS, to support the performance of PSF 1.2.2.2.4.

The NRC staff notes that while the scope of the PDCs discussed in the PSAR is limited to the system performance in AOOs, the performance of NSRST PSFs in the DBE and BDBE regions will be ensured through consideration of the full scope of LBEs when determining special treatments and reliability and capability targets, as stated in PSAR section 7.2 and discussed in section 5.5 of the SE.

Based on the review of information provided, the NRC staff determined that the preliminary SSS design is consistent with the heat removal function listed in PSAR section 7.3.4.1.

Control Water/Steam Ingress Function

PSAR section 7.3.4.1 states that the SSS is designed to meet the following PSF by controlling water and steam ingress during certain LBEs:

- PSF 1.3.2 – remove water/steam following isolation of water/steam source

The CDC for PSF 1.3.2.1 precludes lifting of primary relief valves during certain LBEs that involve water/steam ingress and ensure that SARRDLs are not exceeded. There is not a specific PDC identified that is associated with this PSF. Actuation signals processed by the IPS ensure performance of the automatic isolation functions associated with PSF 1.3.2.1 in the event of a SG tube leak/rupture. No operator actions are needed.

Based on the information provided above, the NRC staff determined that the preliminary SSS design is consistent with the control water/steam ingress function listed in PSAR section 7.3.4.1.

Special Treatments

PSAR section 7.3.4 states that special treatments for the SSS will be identified and selected following expansion of the design and PRA to align with the development of system-level reliability and capability targets, consistent with the process described in NEI 18-04, as endorsed by RG 1.233. A general description of special treatments applied to NSRST SSCs is provided in PSAR section 7.2.

The NRC staff's review finds that while LME has not identified final special treatments for the SSS components, preliminary special treatments are provided through application of the QAP and the selection of codes and standards under PDC 1 as well as through the programmatic special treatments applied for compliance with PDC 6. Additional preliminary special treatments under consideration for the SSS include 100 percent volumetric examination (UT or RT), pressure boundary fusion welds, and full penetration welds without permanent backing strips. To address high-temperature excursions in the SSS components, special treatments may include evaluation of creep-fatigue effects on component cyclic life and demonstration that the cumulative elevated-temperature service hours bound the time basis for B31.1 allowable stress values or an adjustment to the allowable stresses accordingly.

The programs described in PSAR chapter 8 and chapter 12, "Initial Test Program," are identified in the PSAR as programmatic special treatments to be applied to safety-significant SSCs, which includes the SSS.

Based on the review of the functions of the SSS against the description of the codes and standards and qualification programs described in the PSAR, the NRC staff determined that the preliminary special treatments description and commitments are appropriate for the PSAR stage. A final determination of sufficiency of special treatments applied to the SSS will be performed at the OL application stage.

FOAK Testing

Criterion XI, "Test Control," of 10 CFR Part 50, Appendix B "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," 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 SSS components as incorporating FOAK technology, design, or configuration and thus did not identify any FOAK special treatment tests to demonstrate capability of the SSS in performing the necessary NSRST PSF or meeting regulatory requirements.

As part of the regulatory audit, the NRC staff requested the basis for why certain aspects of the SSS design were not identified as FOAK. The NRC staff's concern was that while individual components of the SSS may not be FOAK in nature, their arrangements within the system and the functions performed may be FOAK.

LME, through the audit, provided the following information:

- The Xe-100 SSS design is similar to the thorium high-temperature reactor that operated in Germany. LME further stated that it plans to verify via operational testing that the Xe-100 specific arrangement functions as intended to support reactor modules in all four modes of operation.

- The SSS design and material properties selection will be performed to ensure that they are consistent with the environment in which the system is required to function to support the NSRST PSFs for certain LBEs and normal operations.

The NRC staff reviewed the information provided in the audit to confirm the adequacy of the PSAR statements with regard to FOAK testing. Based on the comparison of the design information in PSAR and the publicly available information, the NRC staff concluded that the components and design features of the SSS do not warrant FOAK special treatment.

The NRC staff will evaluate the continued validity of the FOAK treatment at the OL application when the detailed design information for the SSS is available.

7.3.4.2 Conclusion

Based on the review described above, the NRC staff concludes that the information in PSAR section 7.3.4 is sufficient and meets the applicable guidance and regulatory requirements identified in this SE section and at the beginning of this chapter for the issuance of a CP. Specifically, the information provided:

- meets the requirements of 10 CFR 50.34(a)(3)(i) for the SSS by identifying the applicable PDCs for the preliminary design,

- meets the requirements of 10 CFR 50.34(a)(3)(ii) by describing how the SSS design satisfies the applicable PDCs, and

- meets the requirements of 10 CFR 50.34(a)(3)(iii) by providing the materials of construction and general arrangement of the SSS where available.

7.3.5 Helium Circulator System

PSAR section 7.3.5 describes the HCS. The HCS performs a critical role in maintaining primary coolant circulation under normal, startup, and shutdown conditions, supporting heat removal and plant stability. The HCS circulates helium through the primary system to transfer heat from the reactor core to the SG for power production. The HCS is part of the HPB, and it functions to maintain primary system pressure. The HCS includes:

- Helium circulator: A blower system driven by an electric motor with electromagnetic bearings (and backup auxiliary bearings), a circulator cooling system, and assembly support structure.

- Variable frequency drive (VFD): An alternating current variable speed drive that controls circulator motor speed. There is one VFD for each circulator.

- Electromagnetic bearing controller (EMBC): Regulates rotor position relative to the stator by adjusting direct current voltage to the electromagnetic bearings. Each set of electromagnetic bearings is controlled by one EMBC.

- Circulator inlet assembly: The subsystem interfaces with the SG and with the circulator blower inlet and facilitates helium flow from the SG into the circulator.

- Helium circulator system controller (HCSC): Interfaces with the plant control and data acquisition system (PCDAS) to manage circulator controls, monitor temperatures and vibrations, and protect equipment. Each circulator is controlled by one HCSC.

The HCS interfaces with the following SSCs:

- Steam generator system (SGS): structural and fluidic interface with SGS,

- Helium service system (HSS): interface via helium isolation valve for supply of purified helium,

- Nuclear island cooling water system (NICW): interface with the RB cooling water system (RBCW) via the cooling jacket water inlet and outlet connectors,

- Nuclear island electrical system (NIE): interface via connectors to the VFD, EMBC, and HCSC for highly reliable power to the motor, electromagnetic bearings, and controllers,

- Nuclear instrumentation system (NIS): interface via the DCS and RPS to sense condition requiring HCS response,

- Distributed control system (DCS): interface via the HCSC to provide speed setpoints to the HCSC and receive speed feedback signals from the VFD.

The industry standard codes and materials used to construct the HCS are provided in PSAR table 7.3.5-1 and the process flow chart for SSS is provided in PSAR figure 7.3.5-1.

The system-specific regulatory requirements HCS are as follows:

- 10 CR 50.34(a)(5); and

- 10 CFR 50, Appendix B, Criterion XI

The system-specific guidance for evaluation of the HCS is as follows:

- DANU-ISG-2023-01 “Material Compatibility for non-Light Water Reactors,” (ML22203A175);

- RG 1.87 Revision 3, “Acceptability of ASME Code Section III, Division 5, ‘High Temperature Reactors,’” (ML25176A084); and

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

In addition to PDCs 1-6, the following PDC are applicable to the HCS:

- 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-CDC 34- Residual heat removal

7.3.5.1 Technical Evaluation

Safety Significance

The HCS along with the associated subsystems is classified as NSRST. The HCS is relied upon to demonstrate adequate DID to restore active heat removal following spurious circulator trips, and to provide sufficient heat removal from the core during LBEs which end in SSS cooling. The PSFs performed by the HCS for DID are listed in PSAR table 5.5-7. The HCS is designed to:

- provide an active means of transfer of fission product decay heat and other residual heat from the reactor core to ultimate heat sink at a rate that SARRDLs are not exceeded during AOOs, and

- maintain HPB integrity by controlling the frequency of depressurization events associated with different break sizes.

The PSAR lists the NSRST PSFs supported by the HCS to demonstrate adequate DID. As confirmed by the NRC staff via RCI (ML26239A157), LME is incorporating the results of updated risk-significance evaluations following the NEI 18-04 methodology. The preliminary results of these evaluations indicate that the HCS is risk significant, which would mean that the failure of the prevention or mitigation function of the SSS could lead some of the LBEs crediting the HCS to exceed the F-C target.

The PSAR identifies portions of the HCS as NSRST with the overall system being categorized as NSRST. The PSAR did not identify any NST components in the HCS system. In section 5.5.1 of the SE, the NRC staff determined that the classification of HCS as an NSRST system 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)

All HCS components are identified as NSRST and will be designed to the industry consensus codes and standards as discussed in PSAR section 3.8 as well as listed in PSAR table 7.3.5-1. The components ensuring sufficiency of structural integrity of the HCS are listed in PSAR

table 7.3.5-2 and all other HCS components are listed in PSAR table 7.3.5-3. The use of ASME BPVC section III, division 5 for major components within the HCS is a special treatment selected to provide additional assurance that the HCS can achieve its safety functions. The PSAR states that none of the HCS SSCs operate at or above the elevated-temperature conditions defined in ASME section III.5, table HAA-1130-1. As stated in PSAR tables 7.3.5-2 and 7.3.5-3, some of the materials, codes of construction, and design conditions have not been determined for the HCS components. The NRC staff finds the design conditions, materials, construction codes, and special treatments for the preliminary design of the HCS to be appropriate for the CP application stage. The NRC staff will review the detailed design and construction specification for the final design of the HCS at the OL application stage.

The NRC staff reviewed information in the PSAR which identifies a dissimilar metal joint at the circulator pressure vessel closure end and cylinder. The joint in question is between austenitic stainless steel and ferritic steel and the PSAR states that this joint will operate at low-temperature service conditions as defined in ASME BPVC III.5, table HAA-1130-1. The circulator pressure vessel is constructed in accordance with ASME BPVC III.5-2023 and RG 1.87, which require the dissimilar metal weld to be qualified in accordance with ASME BPVC, section IX.

The methodology for the structural evaluation of the metallic SSCs is provided in PSAR section 3.8, and the NRC staff's evaluation is discussed in section 3.8.1 of this SE. The PSAR states that the metallic components listed in PSAR table 7.3.5-1 operate at low-temperature design conditions (under normal, standby operation) and thus their structural failure modes are adequately addressed by the use of codes such as ASME BPVC section III.5, where identified.

Based on the information provided, the NRC staff determined that the preliminary design of the HCS is consistent with PDC 1.

Design Basis for Protection Against Natural Phenomena (PDC 2)

The HCS components are protected against natural phenomena by their location within the SST. Specifically, the SST provides protection from applicable natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions. The adequacy of the SST to protect the safety-significant SSCs is evaluated in section 6.4.1 of this SE. The HCS components within the SST will be designed to be protected from 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, the SR structures are designed to withstand loads resulting from natural phenomena, including seismic events, tornadoes and associated missiles, external flooding, and extreme climate conditions. PSAR section 6.4.1.2.2 states that the openings in the SST prevent water intrusion by being located above the flood height or by being designed to be substantially impermeable. Moreover, the NRC staff's review determined that the HCS location within SST is higher than the site's still water flood elevation and therefore any water ingress into the structure would not be significant enough to pose a flooding hazard for the HCS. The NSRST HCS SSCs are designed to withstand seismic events and achieve performance levels corresponding with their designated category, as shown in PSAR table 3.7-2 and described in PSAR section 3.7. The NRC staff's evaluation of the seismic design basis is presented in section 3.7 of this SE.

Based on the information provided above, the NRC staff determines that the preliminary design of the HCS is consistent with PDC 2.

Fire Protection (PDC 3)

The HCS components will be constructed using noncombustible or fire-resistant materials to the extent practicable. The HCS components are located in the SST on top of the SG where fixed combustible materials are negligible and transient combustibles are administratively controlled. PSAR section 8.6 provides a description of the FPP, including a list of program elements that are implemented in an integrated manner to meet the requirements of 10 CFR 50.48(a) as well as PDC 3 to minimize the probability and effect of fires and explosions. These elements include, in part, 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. PSAR section 8.6 also indicates that the FPP elements will be described in the operational FPP to be developed in conjunction with and described in the application for an OL. Moreover, the PSAR section states that the operational FPP will be designed in full conformance with the portions of RG 1.189 that are technically relevant to the design. The evaluation of the FPP is described in section 8.6 of this SE.

Based on the review of the preliminary design features of the HCS and the programmatic commitments described in PSAR section 8.6, the NRC staff finds that the fire protection measures for the HCS are consistent with the intent 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 7.3.5.1, which states that an environmental compatibility assessment is performed using the structural evaluation methodology described in PSAR section 3.8. The acceptability of the methodology is described in section 3.8 of this SE. Further, DBHL related requirements, as described in PSAR section 6.1.3 may be applied to NSRST HCS components to prevent any interactions with SR SSCs that would adversely impact the ability of that SSC to perform its RSFs. The PSAR indicates that the HCS is appropriately protected, commensurate with its safety functions against the dynamic effects of co-located equipment failure within the reactor cavity and SST. The dynamic effects from these co-located systems may include pipe whip, impingement of discharge fluids, generation of missiles, and rapid compartment pressurization.

The NRC staff reviewed the PSAR statements that hazard protection strategies, processes, and methods are part of detailed design considerations which incorporate insights from more detailed deterministic and probabilistic hazard assessments in the FSAR, and that the analysis and design approach to demonstrate that dynamic effects are appropriately and systematically addressed will be presented at the OL application stage.

Based on the review of the information provided, the NRC staff concludes that there is sufficient detail in the PSAR to demonstrate that the HCS is designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, AOOs, DBEs, and DBAs commensurate with the safety significance of the functions to be performed and is consistent with PDC 4.

Sharing of Structures, Systems, and Components (PDC 5)

The NSRST HCS components are not shared between the reactor modules. Each reactor module has its own dedicated HCS. However, the PSAR states that the HCS components share

a foundation system with multiple safety-significant SSCs. The adherence to PDC 5 within the context of shared foundation system is described in PSAR section 3.7.6.3 and the acceptability is documented in section 3.7 of this SE.

Based on the review of the information provided, the NRC staff concludes that there is sufficient detail in the PSAR regarding the design of the preliminary HCS 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 HCS to perform its safety functions, commensurate with the NSRST functions to be performed.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR section 7.3.5, which states that periodic inspections of the passive components are performed in accordance with the RAM program, which is evaluated in SE section 8.10. The HCS design provides access for implementing remote preservice and inservice requirements. The NRC staff reviewed the identified potential MANDE requirements, including a visual examination and secondary-water-side chemistry monitoring. Because the design provides access and proposed monitoring and inspection addresses credible degradation mechanisms, the NRC staff finds that the preliminary inspection and monitoring measures provide reasonable assurance that the functional capability of the HCS will be maintained.

LME states that functional testing of the active components will be performed in accordance with the maintenance program which is evaluated in SE section 8.3. However, as discussed in the safety significance subsection above, LME is in the process of incorporating the results of updated risk-significance evaluations following the NEI 18-04 methodology. The preliminary results of these evaluations indicate that the HCS is risk significant. If the active functions of the SSS are risk significant, based on the scope of the IST program as described in PSAR section 8.4, the risk-significant SSCs would be expected to be included in the IST program. For non-risk-significant active SSCs, the NRC staff notes that the maintenance program does not specify a method for developing and implementing an IST program, such as ASME OM-2. The NRC staff will evaluate the operational program scoping of SSCs during the OL application based on final risk-significance determinations. The NRC staff reviewed potential IST requirements, including exercise testing and diagnostic testing. The planned functional testing is expected to encompass the conditions and ranges associated with operation of the HCS SSCs. The adequacy of the test plan, including test conditions, measurements, and monitoring, will be evaluated during the OL review. In addition, LME states that the preoperational testing, as described in section 12.1.1 and evaluated in SE section 12.1, will be performed to provide a baseline for the inservice inspections.

Based on the discussion above, the NRC staff finds that the risk-significant active SSCs are consistent with PDC 6. For the DID NSRST 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 DID adequacy.

Reactor Design (PDC 10)

The NRC staff reviewed section 7.3.5 which states that PDC 10 is a higher-level requirement that is met for the HCS via application of PDC-CDC 34 and system-level CDCs aligned with NSRST PSF 1.2.2, maintain core heat removal through active means, and NSRST PSF 1.2.2.2, restore active heat removal. The PSAR states that the HCS is designed with appropriate margin to ensure it performs the required PSFs through:

- appropriately designated quality, seismic, and ASME code classifications,
- application of approved codes and standards commensurate with NSRST PSFs applied to HCS, and
- selection of materials with consideration for operating conditions that the HCS components would experience during normal operations and certain LBEs.

The NRC staff reviewed the details of the HCS operation and performance provided in PSAR section 7.3.5.2 and finds it to be acceptable because the elements of the design meet the PSFs applied to the HCS. The acceptability of the elements to meet the PSFs applied to the HCS are discussed in the PDC-CDC 34 description below.

Based on the review of the information provided, the NRC staff concludes that the PDC 10 requirements for the HCS are satisfied by meeting PDC-CDC 34 and the system-level CDCs align with NSRST PSFs 1.2.2 and 1.2.2.2.

Instrumentation and Control (PDC-CDC 13)

The NRC staff reviewed PSAR section 7.3.5, which states that the I&C provided by the DCS system facilitate the performance of NSRST PSFs for the HCS during normal operation and certain LBEs. The PSAR states that I&C are provided to monitor variables over their anticipated ranges during normal operation and certain LBEs and maintain those variables within the prescribed ranges.

Based on the description of the I&C provided for the HCS and their 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 HCS is consistent with PDC-CDC 13.

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

The NRC staff reviewed PSAR section 7.3.5, which states that the pressure retaining part of the HCS supports PDC-CDCs 14, 15 and 16 through the use of appropriate ASME codes and standards as the primary means for assuring HPB integrity and supports 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 the HPB material for the HCS considers operating design conditions, as shown in PSAR tables 7.3.5-2 and 7.3.5-3. In addition, the materials for the HCS that function as part of HPB are selected from ASME BPVC section III.5, with consideration for RG 1.87 exceptions and limitations. The PSAR further states that uncertainties in material properties are considered as part of the methodology described in

PSAR section 3.8 and the acceptability of the methodology is presented in section 3.8 of this SE.

Based on the review of the description of the industry codes and standards used in the HCS design, as discussed in the evaluation of PDC 1 above, and the description of the design philosophy for the pressure retaining part of the HCS, the NRC staff finds that the preliminary design of the HCS is consistent with PDC-CDCs 14, 15 and 16.

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

The NRC staff reviewed the PSAR section 7.3.5.1, which states that the HCS components that function as part of the HPB are designed to maintain integrity during certain LBEs. The PSAR further states that the features that support HPB leakage detection are primarily those associated with the design of the plant sensors and associated I&C systems which monitor key parameters to facilitate detection and isolation of any helium leakage from HPB during certain LBEs. Further, the IPS, as described in PSAR section 7.3.11, is designed to automatically isolate helium sources when such conditions are detected by the NIS (PSAR section 6.4.14). While the scope of the PDCs, as discussed here, is limited to the system performance in AOOs, the performance of NSRST PSFs in the DBE and BDBE region will be ensured through consideration of the full scope of LBEs when determining special treatments and reliability and capability targets, as stated in PSAR section 7.2 and discussed in section 5.5 of the SE.

The NRC staff reviewed PSAR section 7.3.5.3, which details HCS qualification and testing. The PSAR states that special treatment requirements for the HCS will be identified and selected following expansion of the design and PRA to align with the development of system-level reliability and capability targets. The section further states that the selection of codes and standards supports both the plant capabilities for DID and the activities that contribute to programmatic DID. PSAR tables 7.3.5-2 and 7.3.5-3 list industry codes and standards that are preliminarily identified as applicable to the HCS and will inherently influence a baseline level of special treatment requirements. The NRC staff finds that deferring the special treatment requirements for the HCS to the OL application stage following the expansion of the PRA to be acceptable since the use of ASME BPVC section III, division 5 for major components within the HCS, as stated in PSAR, is a special treatment that provides additional assurance that the HCS can achieve its safety functions. Based on the review of the description provided in the PSAR regarding the codes and standards and design features of the HCS, the NRC staff finds that the preliminary design of the HCS is consistent with PDC-CDC 30.

Fracture Prevention of Reactor Helium Pressure Boundary (PDC 31)

The NRC staff reviewed PSAR section 7.3.5.1, which states that the design of the HCS components that function as part of the HPB, reflect consideration of material properties and service conditions during normal operations and certain LBEs. For the HCS, PDC 31 is met by using ASME BPVC section III.5, as endorsed in RG 1.87, Revision 3 to design and construct components of the HCS that are part of the HPB. The HCS is classified as SC-IIA as described in PSAR table 3.7-3. The NRC staff's evaluation of the seismic design basis is presented in section 3.7 of this SE.

The NRC staff finds that the HCS preliminary design is sufficient to maintain the HPB integrity because RG 1.87, Revision 3 is used to design and construct components of the HCS that are

part of the HPB, and the appropriate seismic category classification is assigned to the HCS. Therefore, the NRC staff finds that the HCS design is consistent with PDC 31.

Residual Heat Removal (PDC-CDC 34)

The NRC staff reviewed PSAR section 7.3.5.1, which states that the HCS is designed to provide an active means to transfer fission product decay heat and other residual heat from the reactor core to a heat sink at a rate such that SARRDLs are not exceeded during AOOs. The HCS provides primary cooling capability during normal operations and certain LBEs. The PSAR states that the HCS is designed with appropriate margin to ensure it performs PSFs 1.2.2 and 1.2.2.2 through:

- appropriately designated quality, seismic, and ASME code classifications,
- application of approved codes and standards commensurate with NSRST PSFs applied to HCS, and
- selection of materials with consideration for operating conditions that the HCS components would experience during normal operations and certain LBEs.

Based on the review of the information provided regarding the selection of appropriate codes and standards, quality, testing, and seismic categorization for the design of the HCS as discussed above for PDC 1-6, the NRC staff concludes that the PDC 34 requirements for the HCS are satisfied by meeting the system-level CDCs that are aligned with NSRST PSFs 1.2.2 and 1.2.2.2.

Safety Functions

Heat Removal Function

The NRC staff reviewed PSAR section 7.3.5.1, which states that the HCS is designed to meet the following PSF through thermal transportation capabilities during normal operations and certain LBEs:

- PSF 1.2.2 – maintain core heat removal through active means
  - PSF 1.2.2.2 – restore active heat removal

The HCS is designed to provide an active means of heat transfer from the primary system to the secondary system. The HCS performance and operation description provided in PSAR section 7.3.5.2 provides information on the active heat removal capabilities of the circulators during normal operations as well as certain LBEs. The controls, logic circuits, and interlocks provided by the DCS allow the HCS to be automatically operated during normal operations and certain LBEs to support NSRST PSFs 1.2.2.2.1, 1.2.2.2.2, or 1.2.2.2.3 related to active heat removal. The HCS is designed to restart and restore forced circulation if the system parameters are within permissible ranges. Operator actions and operator controls are required for the HCS to support NSRST PSF 1.2.2.2.4 to restore active heat removal during certain LBEs that involve water/steam ingress.

Based on the discussion above, the NRC staff finds that the preliminary HCS design is consistent with the heat removal functions.

Radionuclide Retention Functions

The NRC staff reviewed PSAR section 7.3.5.1, which states that the HCS is designed to meet the following PSFs by maintaining the HPB integrity of the HCS:

- PSF 2 – retain radionuclides in the HPB
  - PSF 2.7 - maintain HPB pressure integrity

The HCS components that function as part of the HPB are designed to maintain integrity during normal operation and certain LBEs. Further, the HCS qualification and testing, discussed in the PSAR, states that special treatment requirements for the HCS will be identified and selected following expansion of the design and PRA to align with the development of system-level reliability and capability targets. This is acceptable since the PSAR states that ASME BPVC section III.5, as endorsed in RG 1.87, Revision 3, will be used to design and construct components of the HCS that are part of the HPB. The design description of the HCS provided details on how the system maintains primary helium flow rate and pressure to circulate helium through the primary system while minimizing helium leakage between system interfaces. The PSAR states that no operator actions, logic circuits, or interlocks are needed for the HCS to support NSRST PSFs 2.7.1, 2.7.2, or 2.7.3.

Based on the discussion above, which states that the HPB parts of the HCS are designed to maintain integrity during normal operation and certain LBEs, the NRC staff finds that the preliminary HCS design is consistent with the radionuclide retention function.

The NRC staff notes that while the scope of the PDCs as discussed in the PSAR is limited to the system performance in AOOs, the performance of NSRST PSFs in the DBE and BDBE region will be ensured through consideration of the full scope of LBEs when determining special treatments and reliability and capability targets, as stated in PSAR section 7.2 and discussed in section 5.5 of the SE.

Special Treatments

PSAR section 7.3.5 states that special treatments for the HCS will be identified and selected following expansion of the design and PRA to align with the development of system-level reliability and capability targets, consistent with the process described in NEI 18-04, as endorsed by RG 1.233. A general description of special treatments applied to NSRST SSCs is provided in PSAR section 7.2.

The NRC staff's review finds that while LME has not identified final special treatments for HCS components, preliminary special treatments are provided through application of the QAP and the selection of codes and standards under PDC 1 as well as through the programmatic special treatments applied for compliance with PDC 6. The selection of codes and standards inherently supports a baseline level of special treatments for the HCS.

The programs described in PSAR chapter 8 and chapter 12 are preliminarily identified in the PSAR as programmatic special treatments to be applied to safety-significant SSCs.

Based on its review of the functions of the HCS against the description of the codes and standards and qualification programs described in the PSAR, the NRC staff determined that the preliminary special treatments description and commitments are appropriate for the PSAR stage.

FOAK Testing

Criterion XI, "Test Control," of 10 CFR Part 50, Appendix B 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."

At the preliminary stage of the HCS design, the PSAR identified several HCS components that are deemed to be incorporating FOAK technology, design, or configuration and thus require FOAK treatment. A list of FOAK special treatment and other performance tests planned for HCS at the preliminary stage are provided PSAR section 7.3.5.

The NRC staff reviewed the information provided for the FOAK treatment for preliminary design of the HCS and concluded that LME appropriately selected testing for the HCS performance consistent with the requirements of 10 CFR Part 50.

Technical Specifications

The HCS has been identified to have technical specifications (TS) based on the preliminary design and the associated preliminary safety analysis, with additional details provided in Part IV, "Proposed Technical Specifications." The proposed list of preliminary system subject to limiting conditions for operation (LCOs) identified in the proposed TS lists the minimum number of circulators available and helium to feedwater mass flow ratio as LCOs to support PSF 1.1.2 and PSF 2.7.

Based on the review of the information provided in the PSAR, the NRC staff finds that the information provided is compliant with 10 CFR 50.34(a)(5) for the HCS given the variables, conditions, or items that are likely to become subjects of TS for the HCS are identified.

7.3.5.2 Conclusion

Based on the review described above, the NRC staff concludes that the information in PSAR section 7.3.5 is sufficient and meets the applicable guidance and regulatory requirements identified in this SE section and at the beginning of this chapter for the issuance of a CP. Specifically, the information provided:

- meets the requirements of 10 CFR 50.34(a)(3)(i) for the HCS by describing the preliminary design identifying the applicable PDCs,

- meets the requirements of 10 CFR 50.34(a)(3)(ii) by describing how the HCS design satisfies the applicable PDCs,

- meets the requirements of 10 CFR 50.34(a)(3)(iii) by providing the materials of construction and general arrangement of the HCS, where available,

- meets the requirements of 10 CFR 50.34(a)(5) by identifying preliminary HCS variables and conditions are subjects of TS, and

- meets the requirements of 10 CFR 50, Appendix B, Criterion XI, “Test Control” by identifying FOAK treatment for preliminary design of the HCS.

7.3.6 Fuel Handling System

Section 7.3.6 of the PSAR describes the FHS. The FHS is connected to the RPV at two locations, one at the top of the RPV where fuel pebbles are loaded into the reactor via the reactor inlet unit (RIU) and another at the bottom of the RPV where the fuel pebbles are removed from the defuel chute pressure vessel (DCPV), which is connected to the bottom of the RPV, via two sphere unloader machines (SUMs). The FHS conveys the fuel pebbles by means of gravitational and pneumatic forces using helium as a transport gas supplied from the HSS. As described in PSAR section 3.10.2, approximately 500 pebbles are in the FHS at any given time. The FHS forms a loop between the SUMs and the RIU that is normally open to the RPV and operates at a pressure slightly higher than reactor pressure to provide a constant flow of helium to the reactor.

The FHS consists of pebble conveying piping, a gas management subsystem connected to the conveying piping, and valve blocks, which are constructed from bored solid metal forgings with inserted pebble-processing components. The conveying piping provides the connections between valve blocks and connects the RIU to the RPV. The gas management subsystem supports movement of pebbles within the FHS, purging graphite dust and contaminants, and pebble movement into and out of the pressurized portion of the FHS. Examples of valve blocks include the RIU and the SUMs. Other valve blocks classify pebbles based on burnup measurement, sort pebbles, transfer pebbles from atmospheric loading stations to the pressurized FHS, and discharge pebbles from the FHS to atmospheric handling in the CPF.

PSAR table 7.3.6-4 shows that select components of the SUMs, including the pressure boundary formed by the body, and the conveying gas filter units are constructed to ASME VIII, division 2. Other component valve block bodies, valve block inserts, and conveying piping are constructed to ASME B31.1.

The majority of the FHS components are in the NSRST FHAB. The SUMs, SUM supporting equipment, and the portion of the pebble conveying piping connected to the RPV and downstream of the RIU are in the NSRST SST. The discharge docking gates, used sphere loading station assemblies, support equipment, and associated piping are in the NSRST CPF.

The system-specific regulatory requirement applicable to FHS is as follows:

- 10 CFR 50.34(a)(5).

In addition to PDCs 1-6, the following PDCs are applicable to the FHS:

- 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 60 - Control of releases of radioactive materials to the environment
- PDC-CDC 61 - Fuel storage and handling and radioactivity control
- PDC 62 - Prevention of criticality in fuel storage and handling
- PDC 63 -Monitoring radioactive releases

7.3.6.1 Technical Evaluation

Safety Significance

The FHS interfaces with each reactor module HPB to control the loading and unloading of fuel pebbles from each module. The interface with the HPB means that the integrity of the FHS pressure boundary affects the reactor safety functions that rely on an intact boundary, including circulation of pressurized helium for active heat removal and maintenance of the HPB as a part of the functional containment. The ability to detect breaks and isolate much of the FHS from the reactor HPB helps to mitigate the safety significance of an FHS pressure boundary loss by supporting the re-establishment of the HPB functional containment and maintenance or restoration of active heat removal.

The FHS pressure boundary integrity also affects the loading and unloading of fuel pebbles from each reactor module. Maintenance of the FHS pressure boundary is necessary to support movement of fuel pebbles through the FHS by gravity and pneumatic forces due to the high normal operating pressure. Failure of the pressure boundary can both degrade the HPB function as a radionuclide retention barrier and challenge the integrity of fuel pebbles by increasing the kinetic energy of affected pebbles. The safety significance of a loss of FHS pressure boundary integrity varies based on the size and location of the break. Small breaks can interrupt the conveyance of fuel or accelerate pebbles within the FHS conveyance piping such that collisions may degrade their radionuclide retention function. Large breaks can result in ejection of pebbles and other dynamic effects. Because each reactor module is designed to deliver fuel pebbles to the SUMs by gravity for fuel unloading, a large rupture affecting the SUM pressure boundary establishes an upper limit on consequences because the rupture potentially allows the entire core inventory to move through the SUM and exit into the reactor cavity.

Other events could challenge safety by affecting the fuel pebble directly, changing how fuel is sorted and routed, or changing the rate of defueling. The method and design of fuel pebble conveyance may mechanically challenge the fuel pebble radionuclide retention capability. The burnup monitoring system used to classify fuel pebbles and direct their movement could misdirect fuel, which could result in exceeding individual pebble burnup limits or change the core loading or canister loading over time. Changes in the rate of defueling can thermally challenge components by significantly increasing the decay heat load that components holding many pebbles encounter.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR section 7.3.6, which states that the non-nuclear codes, ASME section VIII.2 and ASME B31.1, adequately address all structural failure modes with appropriate margins for the FHS commensurate with their safety significance and the low design temperature of less than 300°C. PSAR section 7.3.6 also states that the potential need for enhancements to non-nuclear codes is being evaluated because the FHS components meet the risk-significance criteria. LME is considering the need for special treatments of enhanced weld integrity measures that are more stringent than baseline requirements, such as requiring 100 percent volumetric examination (UT or RT) of pressure boundary fusion welds and requiring full penetration boundary fusion welds. The NRC staff finds the consideration of the identified special treatments appropriate because they address the more likely causes of loss of helium pressure boundary integrity by decreasing the likelihood of defective welds at the completion of construction.

Based on the information above, the NRC staff finds that the preliminary design of the FHS is consistent with PDC 1.

Design Bases for Protection Against Natural Phenomena (PDC 2)

The NRC staff reviewed PSAR section 7.3.6.1.1, which states that the FHS components are located within the SST, FHAB, and CPF, which are described in PSAR sections 6.4.1.2, 7.3.1.1, and 7.3.1.4, respectively. The FHS components are protected against natural phenomena by their location within the NSRST SST, the NSRST FHAB, and the NSRST CPF. Specifically, these NSRST structures are designed to prevent challenges to the SR SSCs that they house. As described in PSAR section 3.7, these structures are designed to withstand loads resulting from natural phenomena, including seismic events, tornadoes and associated missiles, external flooding, and extreme climate conditions. At a minimum, the FHS components within these structures 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, which describe probabilistic assessments of hazards including high winds, wind generated missiles, and external floods to ensure that the SR SSCs will be protected from NSRST FHS failures resulting from those hazards.

The NSRST FHS SSCs are designed to withstand seismic events and achieve performance levels corresponding with their designated SC-IIA, as shown in PSAR table 3.7-2 and described in PSAR section 3.7. This seismic category applies seismic DBHL accelerations to risk-significant portions of the FHS, which includes the SUMs, the RIUs, and the reactor inlet piping that perform the HPB isolation function, and lower accelerations to the remainder of the FHS based on the preliminary assessment of safety significance. The NRC staff's evaluation of the seismic design bases is presented in section 3.7 of this SE.

The PSAR states that hazard protection strategies, processes, and methods are part of detailed design considerations that will incorporate insights from more detailed deterministic and probabilistic hazards assessments in the FSAR. The NRC staff finds this to be reasonable given the preliminary nature of the FHS design. Based on the information provided above regarding the location of NSRST FHS SSCs within structures designed to withstand natural phenomena, the qualification of the FHS SSCs for natural phenomena to ensure their failure does not affect

SR SSCs, and the qualification of the FHS SSCs as SC-IIA, the NRC staff determines that the preliminary design of the FHS is consistent with PDC 2.

Fire Protection (PDC 3)

The NRC staff reviewed PSAR section 7.3.6.1, which states that the FHS is designed and constructed primarily of metallic materials that are inherently noncombustible and non-fire sustaining. The non-metallic materials used (e.g., lubricants and seals) are evaluated for performance in their operating environments, with noncombustible or fire-resistant materials used wherever practical. The NRC staff concludes that the design of the FHS reasonably uses noncombustible or fire-resistant materials to minimize the probability of fire involving the FHS.

PSAR sections 7.3.6.1.2 and 7.3.6.2.2 discuss the management and collection of graphite dust resulting from fuel pebble wear, which the NRC staff evaluated for combustibility if dispersed in air. As evaluated under PDC 60 in this section of the SE, the dust is routinely collected by filters and processed by the gas management subsystem of the FHS, which minimizes the potential for release of graphite dust in the event of a FHS pressure boundary failure. The NRC staff reviewed references addressing graphite dust combustibility, including Electric Power Research Institute (EPRI) Report 1014797, “Graphite Dust Deflagration,” 2007. This report concludes that the risk of a graphite dust explosion during less controlled decommissioning activities at graphite moderated reactors is extremely low due to the properties of graphite. Based on the review of the FHS design presented in the PSAR and information presented in the EPRI report, the NRC staff concludes that failure of the FHS pressure boundary with dispersal of entrained graphite dust into air has negligible potential to lead to a fire or explosion.

Based on the review described above, the NRC staff finds that the design of the FHS is consistent with the criteria of PDC 3 with respect to minimizing the probability and effect of fires and explosions.

Environmental and Dynamic Effects Design Basis (PDC 4)

The NRC staff reviewed PSAR section 7.3.6 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 reviewed in section 3.8 of this SE. Six of the double isolation valves in the FHS are credited with performing a risk-significant function.

The NRC staff reviewed PSAR section 7.3.6 in which LME states that in order to meet PDC 4 requirements of protecting the FHS system against dynamic effects, analyses will be performed to demonstrate adequate protection of the FHS system, including evaluation of the credibility of dynamic hazards from pipe whip, impingement of discharged fluids, generation of missiles and rapid compartment pressurization. Based on the above discussion, the NRC staff finds that the preliminary design of the FHS is consistent with PDC 4.

Sharing of Structures, Systems, and Components (PDC 5)

The NRC staff reviewed PSAR section 7.3.6.1.1, which states that each of the four reactor modules has an independent FHS, with the exception of a section of pebble rolling pathway in

the FHAB and CPF downstream of the discharge lock outlet unit that is shared between two reactor units (1/2 and 3/4).

The shared pebble rolling pathway is not within the HPB, and thus performs no function related to maintenance of the HPB. Due to the low-pressure environment and the gravity rolling operation of the shared section, the NRC staff concludes that there is no excessive challenge to the fuel pebble integrity from mechanical collisions. Additionally, the remainder of the FHS is independent for each reactor module. Based on the review performed, the NRC staff finds that the configuration of the shared pebble rolling pathway outside the HPB does not negatively impact the performance of any FHS safety function and the preliminary FHS design is consistent with PDC 5.

Monitoring, Inspection, Testing, and Surveillance (PDC 6)

The NRC staff reviewed PSAR section 7.3.6, which states that periodic inspections of the safety-significant passive components are performed in accordance with the RAM program, which is evaluated in SE section 8.10. PSAR section 7.3.6.1 states that the FHS design provides access for implementing remote preservice and inservice requirements. The NRC staff reviewed the identified potential MANDE requirements, including a visual examination, health monitoring through temperature and radiation measurements, and helium-side chemistry monitoring and finds them appropriate to provide adequate assurance that the functional capability will be maintained because the potential MANDE elements provide a means of monitoring degradation of passive portions of the FHS commensurate with its safety significance.

Functional testing of the risk-significant FHS active components will be performed in accordance with the IST program. The NRC staff reviewed potential IST requirements, which include exercise, diagnostic, and functional testing, and find them appropriate to maintain the functional capability of the risk-significant, active HPB isolation valves because the IST program is implemented in accordance with ASME OM-2 -2024, as described in PSAR section 8.4 and evaluated in SE section 8.4. In addition, LME states that the preoperational testing as described in PSAR section 12.1.1 and evaluated in section 12.1 of this SE, will be performed to provide a baseline for the inservice inspections.

Based on the discussion above, the NRC staff finds that the testing applied to risk-significant active SSCs is consistent with 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 DID adequacy.

Instrumentation and Control (PDC-CDC 13)

The IPS includes logic circuits and interlocks to control automatic actions needed to perform NSRST PSFs. The NRC staff reviewed PSAR section 7.3.6.2.1, which states that the FHS, in conjunction with the IPS, is designed to automatically isolate the FHS via the RIU and SUM double isolation valves (DIVs) upon detection of an HPB leak. The NRC staff finds that the preliminary FHS design is consistent with PDC-CDC 13 by providing redundant automatically actuated valves to isolate a postulated rupture in the FHS that challenges the HPB and its contribution to functional containment.

Reactor Helium Pressure Boundary (PDC 14); Reactor Helium Pressure Boundary Design (PDC 15); Functional Containment Design (PDC 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 FHS design.

The NRC staff reviewed PSAR section 7.3.6.1.2, which states that these PDC are satisfied by appropriate classification of the FHS as NSRST, use of the approved ASME codes and industrial standards commensurate with the FHS's NSRST functions, and appropriate material selection in combination with the comprehensive material qualification of metallic NSRST SSCs described in PSAR section 3.8.4. These actions serve to minimize the potential for design conditions of the FHS to be exceeded or for material degradation that results in leakage because the evaluation addressed in PSAR section 3.8.4 includes a construction code assessment for material, fabricated geometry, service conditions, and an environmental compatibility assessment.

In PSAR section 7.3.6.1, the applicant describes a preliminary special treatment for risk-significant components of the FHS, which include the SUMs, RIUs, and reactor inlet piping segments. This preliminary special treatment would apply enhanced weld integrity measures more stringent than the baseline requirements of ASME BPVC section VIII and ASME B31.1, which is evaluated under PDC 1 above.

The NRC staff reviewed the applicant's evaluation of the SUM design to support its preliminary classification as NSRST, which the applicant describes in PSAR section 7.3.6.1. The applicant estimates the frequency of a break in the HPB at the SUM that is of sufficient size to prevent pebble bridging and allow continuous discharge of fuel into the reactor cavity would be fully within the BDBE range, considering uncertainties. The applicant states that preliminary quantitative and qualitative evaluations provide reasonable assurance that such a bounding event would have consequences below the LMP 25 rem (0.25 Sv) total effective dose equivalent (TEDE) dose criterion.

The evaluation of this limiting event described in PSAR section 7.3.6.1 considered pebble temperature response, oxidation, and fuel pebble breakage due to ejection under pressure. The applicant states that reactor cavity geometry would not constrain pebble spreading more than assumed in the evaluation and support the rapid cooling of the pebbles in a subcritical configuration. The applicant assumed preferential oxidation of the fuel-free graphite matrix and an oxygen-limited environment, which the applicant states will be subject to confirmation in the final design analysis, and the resulting dose consequences due to oxidation represented less than 1 rem (0.01 Sv). The applicant also states that pebble breakage would be limited by the short time necessary for vessel depressurization and the interference with free pebble discharge due to the dimensions of the HPB rupture such that the number of pebbles broken by forceful ejection would not be sufficient to credibly challenge the LMP 25 rem (0.25 Sv) TEDE dose criterion for high consequence BDBEs. Pebbles discharged under the force of gravity alone are not considered to contribute to dose consequences from mechanical damage. LME states that the above assessment provides reasonable assurance for the CP stage that the frequency and consequences of a large SUM rupture support its NSRST classification. LME also states that the final safety classification and associated analyses, including updated failure modes and effects, PRA, heat removal, oxidation, fuel breakage, criticality, structural, and dose consequence analyses, will be reflected in the FSAR.

The NRC staff assessed the preliminary evaluation with regard to criticality prevention, heat removal adequacy, and continued retention of radionuclides in the fuel pebble. To support this assessment, the NRC staff reviewed the preliminary quantitative analyses in the audit. The NRC staff finds that the applicant's conclusions regarding discharged pebbles settling into a subcritical configuration that supports adequate cooling are reasonable because the size of the RB concrete floor area, as shown in PSAR figures 6.4.1-3 and 6.4.1-5, provides sufficient area for the pebbles to spread into a relatively thin layer no more than a few pebble diameters thick. This pebble configuration results in high-neutron-leakage that prevents criticality. The NRC staff finds that this configuration also provides effective cooling because the bottom layer is in direct contact with the RB base mat and the top layer is cooled by convective and radiative heat transfer to the RCCS. The thin layer of pebbles between the top and bottom layers can reasonably transfer heat by conduction, convection, and radiation to both the top and bottom layers. Radionuclide retention is provided by preventing excessive oxidation of the fuel pebbles. The combination of effective cooling and low oxygen concentration because of the helium release to the reactor cavity support limited oxidation of the fuel. The fuel design supports low doses provided that the oxidation is limited because of the fuel-free outer layers of each pebble that can oxidize without releasing significant quantities of radionuclides.

The NRC staff also considered the potential for mechanical damage to the fuel from the high-energy break. The NRC staff noted that the configuration of the fuel path shown in PSAR figure 6.4.5-1 from the defuel chute, through the DCPV, and into the SUM limits the number of fuel pebbles discharged under pressure because the path to the SUM is relatively long and the directional change in the T-shaped DCPV would reduce the energy in released fuel pebbles. The size of the break and the short time to depressurize combined with the discharge path would limit the amount of fuel discharged under pressure. Consideration of leak-before-break would further reduce the number of pebbles broken under high-pressure rupture scenarios due to the depressurization that occurs before the break reaches a size and shape that permits pebble discharge. Therefore, the NRC staff finds the LME conclusion that only a small fraction of the total reactor fuel inventory would experience discharge under significant pressure that would credibly damage pebbles, to be reasonable. Because fuel breakage is the dominant contributor to dose consequences under conditions limiting fuel oxidation, these considerations are expected to preclude high dose consequences that could occur under higher energy discharge scenarios. In addition, enhancements to the reactor cavity design to ensure low oxidation and reduced pebble breakage appear feasible. Therefore, the NRC staff finds that the dose consequences for SUM pressure boundary failure events of less than the LMP 25 rem (0.25 Sv) TEDE dose criterion for high consequence BDBEs are reasonably assured.

The NRC staff also considered the FHS design for the isolable portion of the system. PSAR section 3.10.2.2 describes an evaluation of an FHS pressure boundary failure permitting discharge of pebbles that could result in mechanical damage. The NRC staff's evaluation in SE section 3.10.1.2, finds that the NSRST FHS design provides sufficient safety margins during normal operations and ensures the frequency of more challenging events is acceptably low.

Considering the above evaluations, the NRC staff finds that, by using FHS design and materials conforming to accepted codes and by incorporating consideration of service conditions, material, the effects of fabricated SSC geometry, and environmental compatibility, the FHS design basis is consistent with the design criteria of PDC 14, PDC-CDC 15, PDC-CDC 16, and PDC 31.

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

The NRC staff reviewed PSAR section 7.3.6.2, which states that the automatic closure of the RIU and SUM DIVs isolates FHS HPB leakage, consistent with the description in the evaluation of PDC-CDC 13 above. The NRC staff finds, consistent with PDC-CDC 30, that the reactor design includes appropriate instrumentation in the IPS to identify HPB leakage, as described in the NRC staff's evaluation of PDC-CDC 13, and to initiate FHS isolation.

Control Release of Radioactive Materials to the Environment (PDC 60)

The NRC staff reviewed PSAR section 7.3.6.1.2, which states that the FHS design uses bellow-sealed valves, welded connections, and double-sealed flanges with leak detection to minimize system leakage. Graphite dust and fission product gases entrained in the FHS helium environment are confined within the FHS pressure boundary. Graphite dust is removed from the continuous conveyance gas flow by the pulse filter unit and collected and removed for processing by the solid waste radwaste handling subsystem. Helium removed during FHS process operations, such as controlled depressurization of FHS piping segments containing fuel indexed for discharge, is routed to the helium recovery system for purification and reuse or filtration and discharge through the nuclear island HVAC (NIHV) system monitored exhaust stack. The NRC staff reviewed the processes included in the FHS design basis for control of radioactive material, and the NRC staff finds that the preliminary FHS design is consistent with PDC 60 for control of radioactive material because the design of the FHS and related gas systems provide for the management of solid radioactive waste and the controlled monitored release of gaseous radionuclides.

Fuel Storage and Handling and Radioactivity Control (PDC 61)

The NRC staff reviewed PSAR section 7.3.6.1.2, which states that shielding is considered in the design and configuration of the FHS to minimize radiological exposures to plant personnel and electronics and PSAR section 7.3.1, which describes the design of the FHAB and CPF to support their shielding functions. The NRC staff reviewed drawings of the FHAB and CPF in PSAR section 7.3.1 and noted that the FHAB and CPF preliminary designs include thick reinforced concrete walls that provide shielding in areas indicated to contain FHS components. The NRC staff notes that the SST provides external shielding and FHS components located within the SST are designed for remote access, as described in PSAR section 7.3.6. Therefore, the NRC staff finds that the preliminary design bases for the FHS and surrounding structures provide shielding, and are consistent with PDC 61.

The confinement function of the FHS can be divided into functions related to the outer pressure boundary of the FHS, control of radionuclides circulating within the FHS, and the radionuclides contained within intact pebbles. PSAR section 7.4.6.1.2 discusses that preliminary analyses indicate that sufficient heat removal is provided by passive conduction from the fuel pebbles to the FHS piping and component metal, ambient air, and the surrounding structure. Documents reviewed by the NRC staff in the regulatory audit indicate that the preliminary design of the FHS contains a small decay heat load of approximately 1 kW distributed among hundreds of fuel pebbles throughout the FHS, and the NRC staff concluded that heat losses to the environment would be adequate by engineering judgment. Also, the NRC staff evaluated thermal challenges to the fuel in SE section 3.10.1.2. Accordingly, the NRC staff finds that the preliminary FHS design provides sufficient residual heat removal that remains available through normal operating

conditions and transients, including DBEs and DBAs, based on the very small heat load and the linear distribution of fuel pebbles within the FHS piping, consistent with PDC 61.

Based on the review performed, the NRC staff finds that the preliminary design of the FHS is consistent with PDC 61 because the preliminary design acceptably addresses heat removal, shielding, containment, confinement, and filtering subsystems.

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

The NRC staff reviewed PSAR section 7.3.6.1.2, which states that the FHS is designed to ensure that fuel pebbles in any potential configuration or handling operation within the FHS remain subcritical. Additionally, the NRC staff reviewed the discussion of criticality safety for the FHS in PSAR section 3.9.6, which indicates that the reasonable assurance that criticality within the FHS is precluded and that the fuel pebble inventory within the FHS is bounded by the used fuel canister analysis showing that 6000 fuel pebbles at an enrichment of 15.5 weight-percent U -235 cannot become critical under any moderation conditions. PSAR section 7.3.6.1.2 describes that the geometrically safe configuration of the FHS inherently prevents criticality within the FHS, and this statement is supported by descriptions of fuel pebbles being maintained in a single row throughout the FHS. Based on the review performed, the NRC staff finds that the preliminary design of the FHS is consistent with PDC 62 due to the geometric configuration of the system and the limited number of fuel pebbles that may cluster together.

Monitoring Fuel and Waste Storage (PDC 63)

The NRC staff reviewed PSAR section 7.3.6.1.2, which states that temperatures are monitored within the FHS to provide indication of a postulated loss of residual heat removal cooling. Additionally, the NIHV system provides temperature monitoring for the buildings that contain FHS components. The radiation monitoring system includes radiation detectors that will detect and notify plant personnel of excessive radiation levels. PSAR section 7.3.6.1.2 also states that alarms and indications are provided to notify operators to initiate appropriate safety actions. Based on the review performed, the NRC staff finds that the preliminary FHS design is consistent with PDC 63 because the FHS contains a small inventory of fuel that does not require active cooling for adequate decay heat removal such that indication of degraded cooling is adequate and that area radiation monitoring provides adequate notice of abnormal conditions.

Safety Functions

PSAR section 7.3.6.1 states that the FHS supports performance of NSRST PSF 2 to retain radionuclides in the HPB, which includes NSRST PSF 2.2 to isolate the HPB and NSRST PSF 2.7 to maintain the HPB pressure integrity. PSF 2.7 encompasses PSF 2.7.1, 2.7.2, and 2.7.3, which involve controlling the frequency of small, medium, and large depressurization events, respectively. In PSAR table 5.3-2, LME identified the following associations between NSRST PSFs and PDC:

- NSRST PSF 2: PDC-CDC 16 (Functional containment design)
- NRST PSF 2.2: PDC-CDC 13 (I&C), PDC-CDC 16 (Functional containment design), and PDC-CDC 17 (Electric power systems)

- NSRST PSF 2.7: PDC-CDC 15 (Reactor helium pressure boundary design), PDC-CDC 16 (Functional containment design), and PDC-CDC 30 (Integrity of reactor helium pressure boundary)

PSAR section 7.3.6.1 includes the following CDC for PSF 2.2.1: “The FHS is designed with the capability of being isolated from the HPB during a depressurization event initiated from the FHS.” PSAR section 7.3.6.2.1 states that this function is performed without operator actions, with the IPS controls and logic circuits performing the isolation function with electrical power support from the NIE.

The bases for classifying SSCs contributing to PSF 2 is addressed in PSAR chapter 4 and PSAR chapter 5, “PRA Safety Function, Design Criteria, and SSC Classification.” PSAR section 5.5.1 describes that the FHS has a risk-significant function, which is identified in PSAR table 5.5-22 as risk significant due to cumulative risk for PSF 2.2.1 to isolate the FHS connection to the reactor HPB. PSAR table 4.2-7 addressing plant capability DID, lists PSF 2.2.1 and PSF 2.7 as performing prevention functions for several small and medium HPB depressurization LBEs. PSAR table 5.5-19 links PSF 2.7 for maintenance of the FHS HPB integrity as providing DID for LBEs associated with HPB depressurization events.

Special Treatments

PSAR section 7.3.6.1.2 describes that the FHS is constructed to the following standards:

- ASME BPVC section VIII, Rules for Construction of Pressure Vessels, Division 2 - Alternative Rules (2023)

- ASME B31.1, Power Piping (2022)

The description in PSAR section 7.3.6.3 states that the special treatments for the FHS will be identified and selected following expansion of the design and PRA to align with the development of system-level reliability and capability targets, consistent with the process described in NEI 18-04, as endorsed by RG 1.233. LME addresses preliminary special treatments in the discussion of PDC 6, which the NRC staff evaluates above.

The programs described in PSAR chapter 8 and chapter 12 are preliminarily identified as programmatic special treatments to be applied to safety-significant SSCs.

The NRC staff reviewed the functions of the FHS against the scope of the qualification, testing, and programs described in the PSAR. The applicable programmatic treatments from PSAR chapters 8 and 12 are considered for safety-significant SSCs. The PSAR states that the NEI 18-04 process for the selection of special treatments is used to determine the extent of the special treatment requirements to NSRST components of the FHS. As described in the review of PDC 6 above, the NRC staff finds that the FHS special treatments commitments are appropriate for the PSAR stage.

PSAR table 7.3.6-5 provides a preliminary description of the material of construction for FHS components. PSAR section 7.3.6.2.1 states that the final selection of certain materials, including bolting, pressure-retaining components, and sealing materials is pending further evaluation and

testing to confirm adequate performance under anticipated environmental conditions during normal operations and LBEs.

First-of-a-Kind Testing

PSAR table 7.3.6-3 provides a listing of FHS components considered FOAK and the preliminary plans for testing to verify performance. The FHS valve blocks, cartridge valve inserts, and the SUM spindles are among those components considered FOAK and that are subject to further evaluation. Proposed testing includes testing of valve assemblies to verify HPB isolation, which is a risk-significant function. Additional component testing is identified to verify the functions related to the PSFs of preventing HPB leaks and preventing excessive challenges to the fuel within the FHS. Other components listed do not perform safety-significant functions and are not within the scope of this evaluation.

The NRC staff reviewed the proposed FOAK testing for FHS components and finds the scope reasonable for confirming performance of the system, consistent with the QA requirements applicable to NSRST components under the program described in PSAR section 8.1.

Technical Specifications

The applicant identified certain FHS functions for TS. The proposed list of preliminary systems subject to LCOs identified in table IV-3 of the proposed TS lists fuel burnup measurement and pebble dimension checks as potential LCOs related to the radionuclide retention function. In addition, in the row addressing the DCPV, table IV-3 lists FHS isolation valve status.

The NRC staff reviewed information provided in the PSAR related to the proposed parameters. The burnup measurement system, which includes detectors and an algorithm to convert energy dependent photon activities to a burnup measurement, ensures that fuel pebbles do not exceed their licensed burnup limits. This parameter may be important to radionuclide retention within the pebbles under accident conditions and potentially serves as an initial condition of safety analyses. The dimensional check is similarly important to verifying pebble capability to retain radionuclides by screening out pebbles that have incurred mechanical damage affecting the potential release characteristics under accident conditions. The NRC staff finds that the information provided is compliant with 10 CFR 50.34(a)(5) given the variables, conditions, or items that are likely to become subjects of TS for the facility are identified.

7.3.6.2 Conclusion

Based on the review described above, the NRC staff concludes that the information in PSAR section 7.3.6 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 the requirements of 10 CFR 50.34(a)(3)(i) for the FHS by describing the preliminary design identifying the applicable PDCs,

- meets the requirements of 10 CFR 50.34(a)(3)(ii) by describing how the FHS design satisfies the applicable PDCs,

- meets the requirements of 10 CFR 50.34(a)(3)(iii) by providing the materials of construction and general arrangement of the FHS, where available, and

- meets the requirements of 10 CFR 50.34(a)(5) by identifying preliminary FHS variables and conditions that are subjects of TS.

7.3.7 Helium Service System

PSAR section 7.3.7 describes the preliminary design of the HSS. The HSS is a collection of subsystems that store, monitor, transfer, and purify the helium used in the primary system and provides a pressure retaining interface with the HPB.

The HSS comprises the following subsystems:

- Helium purification system (HPS), classified as NSRST and discussed further in this section.

- Helium supply and storage system, classified as NST and described in PSAR section 1.1.4.

- Gas evacuation system, classified as NST and described in PSAR section 1.1.4.

- Gas analysis and sampling system, classified as NST and described in PSAR section 1.1.4.

- Helium recovery system, classified as NST and described in PSAR section 1.1.4.

- Liquid nitrogen system, classified as NST and described in PSAR section 1.1.4.

The HPS diverts a portion of primary helium flow from the outlet of the HCS, passing it through a series of filters, heat exchangers, a copper oxide bed, moisture separator, molecular sieve, and low-temperature adsorbers, before returning the purified helium to the inlet of the HCS. The HPS provides a portion of the HPB.

The HPS components are located in the helium storage facility with interfacing piping routed to the FHAB and SST.

The HPS supports the following NSRST PSF, which is required to demonstrate adequate DID for retaining radionuclides in the HPB during specific LBEs:

- PSF 2.7 - maintain the HPB pressure integrity

Interfaces with the following SSCs are needed for HPS to support NSRST PSFs:

- Steam generator system (SGS)
- Helium circulator system (HCS)
- Fuel handling system (FHS)

The HPS system does not support any SR required safety functions.

In addition to PDCs 1-6, the following system-specific PDCs are applicable to the HSS:

- PDC-CDC 13 - Instrumentation and controls
- PDC 14 - Reactor helium pressure boundary
- PDC-CDC 15 - Reactor helium pressure boundary design
- PDC-CDC 16 - Functional containment
- PDC-CDC 30 - Integrity of reactor helium pressure boundary
- PDC 31 - Fracture prevention of reactor helium pressure boundary

7.3.7.1 Technical Evaluation

Safety Significance

The HPS subsystem of the HSS is classified as NSRST. This classification is consistent with the two NSRST PSFs the HPS is credited for performing. The HPS is relied upon to demonstrate adequate DID for maintaining HPB pressure integrity during the LBEs identified in PSAR table 5.5-20. As part of the HPB, the HPS provides a barrier to radionuclide release. Additionally, the HPS can be isolated from the rest of the HPB upon detection of a helium leak via isolation valves contained in the SGS to mitigate the consequences of a HPB leak in the HPS.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR section 7.3.7, which states that all the HPS components are identified as NSRST and will be designed to ASME BPVC section VIII and ASME B31.1, as shown in PSAR table 7.3.7-1.

Due to the low temperature of the system and low consequence of leakage, enhanced weld integrity measures beyond the non-nuclear code requirements are considered unnecessary but the PSAR states that they will be considered as the design matures. The structural evaluation of HPS metallic SSCs is provided in PSAR section 3.8, and the acceptability of the evaluation is discussed in section 3.8 of this SE. The NRC staff finds the selected codes and standards adequate for the preliminary design due to the low-temperature design conditions and that structural failure modes with appropriate margin are adequately addressed by the respective codes and standards listed.

The HPS is designed, fabricated, constructed, and tested in accordance with the QAP described in part III of the LMGS QAPD and PSAR section 8.1. Additional special treatments related to QA 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 QA requirements under PDC 1 for the NSRST portions of the HPS. 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 information provided above, the NRC staff finds that the HPS is consistent with PDC 1.

Design Bases for Protection Against Natural Phenomena (PDC 2)

The NRC staff reviewed PSAR section 7.3.7, which discusses that the HPS is located within the HSF with interfacing piping routed to the FHAB and SST, which protect the HPS components from applicable natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions as described in PSAR section 3.7. As stated in PSAR table 3.7-2, the NSRST portions of the HPS are categorized as SC-IIA for seismic safety significance to ensure they can perform the associated NSRST PSFs during and following a seismic event. The seismic design requirements associated with SC-IIA 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. The PSAR states that hazard protection strategies, processes, and methods are part of detailed design considerations that will incorporate insights from more detailed deterministic and probabilistic hazards assessments in the FSAR. The NRC staff finds this to be reasonable given the preliminary information presented above shows reasonable assurance of protection against natural phenomena without loss of capability of HPS to perform its safety-significant functions.

Based on the information provided above, the NRC staff finds that the preliminary HPS design is consistent with PDC 2.

Fire Protection (PDC 3)

The NRC staff reviewed PSAR section 7.3.7.1, which includes the discussion that the design, location, and configuration of the HPS minimize the probability and effects of fires and explosions and that the fire protection systems are designed, installed, and tested to protect the HPS and other safety-significant SSCs. The HPS components are composed of primarily metallic materials that are high-temperature resistant, inherently noncombustible, and non-fire sustaining. The HPS components are located in the HSF with interfacing piping in the FHAB and SST where fixed combustible materials are negligible and transient combustibles are administratively controlled. PSAR section 8.6 provides a description of the FPP including a list of program elements that are implemented in an integrated manner to meet the requirements of 10 CFR 50.48(a) as well as PDC 3 to minimize the probability and effects of fires and explosions. These elements include, in part, 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. PSAR section 8.6 also indicates that the FPP elements will be described in the operational FPP to be developed in conjunction with and described in the application for an OL. Moreover, the section states that the FPP will be designed in full conformance with the portions of RG 1.189 that are technically relevant to the design. The acceptability of the FPP is described in section 8.6 of this SE.

Based on the preliminary design features of the HPS and the programmatic commitments described in PSAR section 8.6, the NRC staff finds that the fire protection measures for the HPS are consistent with PDC 3 to minimize the probability and effects of fires and explosions.

Environmental and Dynamic Effects Design Basis (PDC 4)

The NRC staff reviewed PSAR section 7.3.7 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 reviewed in section 3.8 of this SE.

Further DBHL related requirements, as described in PSAR section 6.1.3 may be applied to NSRST HPS components to prevent any interactions with SR SSCs that would adversely impact the ability of that SSC to perform its RSFs. The PSAR indicates that the HPS is appropriately protected, commensurate with its safety functions against dynamic effects of co-located equipment failure. The dynamic effects from these co-located systems may include pipe whip, impingement of discharge fluids, generation of missiles, and rapid compartment pressurization.

The PSAR states that hazard protection strategies, processes, and methods are part of detailed design considerations which incorporate insights from more detailed deterministic and probabilistic hazards assessments in the FSAR, and that the analysis and design approach to demonstrate that dynamic effects are appropriately addressed will be presented at the OL application stage.

Based on the information evaluated above, the NRC staff concludes that there is sufficient detail in the PSAR to demonstrate that the preliminary HPS design accommodates the effects of and is compatible with the environmental conditions associated with normal operation, maintenance, testing, AOOs, DBEs, and DBAs commensurate with the safety significance of the functions to be performed and is consistent with PDC 4.

Sharing of Structures, Systems, and Components (PDC 5)

The NRC staff reviewed PSAR section 7.3.7, which includes a description of the NSRST portions of the HSS. The NRC staff finds the general description of LMGS sufficient to determine that the HPS is not shared between multiple reactor modules and therefore is consistent with PDC 5.

PDC 6 - Monitoring, Inspection, Testing, and Surveillance

The NRC staff reviewed PSAR section 7.3.7.1, which states that periodic monitoring of the passive NSRST HPS SSCs will be performed in accordance with the RAM program described in SE section 8.10. The PSAR states that LME will select the appropriate MANDE requirements to apply to HPS from the degradation mechanism assessment performed as part of the RAM program. The potential MANDE requirements for the HPS include visual examination, SSC health monitoring through temperature and radiation measurements, as well as helium-side chemistry monitoring. Because the design provides access and proposed monitoring and inspection addresses credible degradation mechanisms, the NRC staff finds that the preliminary monitoring measures provide reasonable assurance that the functional capability of the HPS will be maintained.

LME states that the startup and preoperational testing, as described in PSAR section 12.1.1 and evaluated in SE section 12.1, will be performed to provide a baseline for the inservice inspections.

The NRC staff will evaluate the RAM program 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 DID adequacy.

PDC-CDC 13 - Instrumentation and Control

The NRC staff reviewed PSAR section 7.3.7.1, which states that instrumentation is designed to monitor HPS pressure, temperature, and other variables over their anticipated ranges during normal operations and LBEs, as appropriate, to support and provide indication of functions performed by safety-significant SSCs. Instrumentation within the HPS interfaces with the IPS to initiate isolation of the HPS via isolation valves contained within the SGS upon detection of a HPB leak. The NRC staff finds that the preliminary design of the HPS is consistent with PDC-CDC 13 based on the information discussed above, including instrumentation to monitor for helium leakage and interfaces with the IPS for leak isolation.

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

The NRC staff reviewed PSAR section 7.3.7, which states that the pressure retaining part of the HPS supports PDC-CDCs 14, 15, 16 and 31 through use of the appropriate ASME codes and standards as the primary means of supporting performance of NSRST PSF 2.7 to 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 HPS considers operating design conditions, as shown in PSAR table 7.3.7-1. The PSAR further states that uncertainties in material properties are considered as part of the methodology described in PSAR section 3.8 and the acceptability of the methodology is presented in section 3.8 of this SE. The HPS is designed to prevent abnormal leakage and unacceptable moisture ingress consistent with system reliability targets, in part, by minimizing leakage points with welded piping connections and hermetically sealed components. The HPS is designed with consideration of service temperatures; service degradation of material properties; fatigue; stress rupture; other conditions of the HPB material under operating, maintenance, testing, AOOs, DBEs, and DBAs; the uncertainties in determining material properties; the effects of irradiation and helium composition, including contaminants and reaction products, on material properties, residual, steady-state, transient stresses; and the size of flaws.

Based on the description of the industry codes and standards used in the HPS design, as evaluated in the discussion of PDC 1 above, and the description of the design philosophy for the pressure retaining part of the HPS, including features to minimize leakage, the NRC staff finds that the preliminary design of the HPS is consistent with PDC 14, PDC-CDC 15, PDC-CDC 16, and PDC 31.

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

The NRC staff reviewed PSAR section 7.3.7, which states that the HPS is designed to detect and, to the extent practical, identify the source of reactor helium leakage. The PSAR also states that when low HPB pressure is detected by the NIS, indicating an HPB leak, the IPS isolates the HPS by closing valves in the SGS to isolate the HPS from the HPB. Based on the review performed, the NRC staff finds that the preliminary design of the HPS is consistent with PDC-CDC 30 because it has the capability to detect helium leakage and to be isolated from the remainder of the HPB.

Safety Functions

PSAR section 7.3.7.1 states the HPS supports the following NSRST PSFs and PSF CDCs:

- PSF 2 - retain radionuclides in the HPB
    - PSF 2.7 - maintain the HPB pressure integrity
        - CDC for PSF 2.7.1 – The HPS is designed to minimize the likelihood of small depressurization AOOs.
        - CDC for PSF 2.7.2 – The HPS is designed to prevent medium depressurization AOOs.
        - CDC for PSF 2.7.3 – The HPS is designed to prevent large depressurization AOOs.

The preliminary design of the HPS supports performance of these functions through the application of PDC-CDC 15 and PDC-CDC 16. The NRC staff's evaluation of these PDC-CDCs is discussed above.

Special Treatments

PSAR section 7.3.7 states that final special treatments for the HPS will be identified and selected following expansion of the design and PRA to align with the development of system-level reliability and capability targets, consistent with the process described in NEI 18-04, as endorsed by RG 1.233. A general description of special treatments applied to NSRST SSCs is described in PSAR section 7.2.

The NRC staff finds that while LME has not identified final special treatments for HPS components, preliminary special treatments are provided through the application of QA under PDC 1 as well as through the programmatic special treatments applied for compliance with PDC 6. Additionally, preliminary special treatments under consideration for HPS system include 100 percent volumetric examination (UT or RT) of pressure boundary fusion welds and requiring full penetration pressure boundary fusion welds.

The programs described in PSAR chapter 8 and chapter 12 are preliminarily identified in the PSAR as programmatic special treatments to be applied to safety-significant SSCs.

Based on its review of the functions of the HPS against the description of the codes and standards and qualification programs described in the PSAR, the NRC staff determined that the preliminary special treatments description and commitments described above are appropriate for the PSAR stage. A final determination of sufficiency of special treatments applied to the HPS system will be performed at the OL application stage.

PSAR section 7.3.7.3 states that at the preliminary stage of design, no HPS component has been identified as incorporating FOAK technology, design, or configurations that would warrant special treatment testing. The NRC staff will evaluate the continued validity of the FOAK

treatment at the OL application stage when the detailed design information for the RBCW is available.

7.3.7.2 Conclusion

Based on the review described above, the NRC staff concludes that the information provided in PSAR section 7.3.7 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and in the beginning of this chapter for the issuance of a CP. Specifically, the information provided:

- meets the requirements of 10 CFR 50.34(a)(3)(i) for the HPS by describing the preliminary design identifying the applicable PDCs,

- meets the requirements of 10 CFR 50.34(a)(3)(ii) by describing how the HPS design satisfies the applicable PDCs, and

- meets the requirements of 10 CFR 50.34(a)(3)(iii) by providing the materials of construction and general arrangement of the HPS, where available.

7.3.8 Nuclear Island Electrical System

PSAR section 7.3.8 describes the NIE. The NIE distributes electrical power to and provides electrical protection for NI electrical loads and is classified as NSRST.

The PSAR states that the NIE comprises the following subsystems:

- NI electric power system: This subsystem includes the equipment that provides both the normal and backup electrical power supplies to the NI.

- NI earthing system: This subsystem includes both the grounding and lightning protection equipment necessary to protect the NIE.

- NI cable & raceway system: This subsystem includes all NIE cabling as well as the equipment necessary for routing of NIE cabling.

- NI building electrical services: This subsystem provides indoor and outdoor lighting as well as non-process power supply, such as convenience outlets, to the NI.

- NI electrical process protection: This subsystem includes electrical equipment needed for both heat trace and cathodic protection of process systems.

PSAR section 7.3.8.1 states that the redundant electrical power sources that supply the NIE include:

- Two physically independent transmission lines connected to the plant switchyard, designed and located to minimize simultaneous failure;

- Two turbine-generator sets; and

- Six standby diesel generators.

PSAR section 7.3.8.1 states the following:

- The NIE includes sufficient independence, redundancy, and testability of both the normal and alternate sources of electrical power to reliably perform system functions.

- The NIE provides both normal and alternate supplies of uninterruptible power to 120 volts alternating current and 125 volts direct current loads with backup power from station batteries or localized uninterruptible power supplies.

The system-specific regulatory requirement applicable to NIE is as follows:

- 10 CFR 50.48(a).

The system-specific guidance for the evaluation of the NIE is as follows:

- RG 1.75, “Criteria for Independence of Electrical Safety Systems”;

- RG 1.93, “Availability of Electric Power Sources”;

- RG 1.212, “Sizing of Large Lead-Acid Storage Batteries”;

- RG 1.128, “Installation Design and Installation of Vented Lead-Acid Batteries for Stationary Applications for Production and Utilization Facilities”; and

- RG 1.129, “Maintenance, Testing, and Replacement of Vented Lead-Acid and Storage Batteries for Production and Utilization Facilities.”

In addition to PDC 1-6, the following PDC are applicable to the NIE:

- PDC 10 - Reactor design
- PDC 17 - Electric power systems

7.3.8.1 Technical Evaluation

Safety Significance

PSAR section 7.3.8.1 states that the NIE is NSRST and the term “safety-significant” for the NIE has the same meaning as NSRST. As confirmed by the NRC staff via RCI (ML26239A157), LME is in the process of incorporating the results of updated risk-significance evaluations following the NEI 18-04 methodology. The preliminary results of these evaluations indicate that the NIE is risk significant, which would mean that failure of prevention or mitigation function of the NIE could lead some of the LBEs crediting the NIE to exceed the F-C target.

As stated in PSAR sections 8.2 and 8.4, SSCs that are risk significant will be included in the reliability and integrity management (RIM) and IST programs and the NRC staff will evaluate the operational program scoping of SSCs during the OL application stage based on the final risk-significance determinations.

Safety Functions

PSAR section 7.3.8.1 states that the NIE supports the following NSRST PSFs and PSF CDCs:

- PSF 1.1.3 - control reactivity with movable poisons
  - CDC for PSF 1.1.3.1 – NIE provides power to NIS, IPS, RCSS and RBCW
- PSF 1.2.2 - maintain core heat removal through active means
  - CDC for PSFs 1.2.2.2.1, 1.2.2.2.2, 1.2.2.2.3 and 1.2.2.2.4– NIE provides power to HCS, SSS, NIS, DCS and RBCW
  - CDC for PSF 1.2.2.3– NIE provides power to RCCS
- PSF 1.3.2 - remove water/steam following isolation of water/steam sources
  - CDC for PSFs 1.3.2.1 – NIE provides power to SGS, SSS, NIS, and IPS
- PSF 2.2 - isolate HPB
  - CDC for PSFs 2.2.1 – NIE provides power to FHS, SGS, NIS and IPS

The preliminary design of the NIE supports performance of these functions through the application of PDC 1-6, PDC 10, and PDC-17. The NRC staff's evaluation of these PDC-CDCs is discussed below.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

For the NIE, all components are identified as NSRST and will be designed to the industry consensus codes and standards listed in PSAR tables 1.4-3 and 7.3.8-2. Special treatments for the NIE, as discussed in PSAR section 7.3.8.3, have not yet been selected, and will be submitted as part of an OL application. The applicant stated in PSAR section 7.3.8.3 that preliminary special treatments are provided in the description of PDC 6 and in the selection of codes and standards. The applicant provided the applicable codes and standards for the NSRST offsite and onsite power systems in PSAR tables 1.4-3 and 7.3.8-2. The NIE is designed, fabricated, constructed, and tested in accordance with the QAP described in part III of the LMGS QAPD and PSAR section 8.1. Additional special treatments related to QA 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 QA requirements under PDC 1 for the NSRST portions of the NIE. The NRC staff's evaluation of the LMGS QAPD is provided in the QAPD TR SE and section 8.1 of this SE.

Since the applicable codes and standards address the design, capacity, capability, independence, maintenance, inspection, and testing of the NSRST NIE, the NRC staff finds that the NIE is consistent with PDC 1.

Design Basis for Protection Against Natural Phenomena (PDC 2)

The NRC staff reviewed PSAR section 7.3.8.1, which states that the NIE: (1) is designed to withstand seismic events and achieve performance levels corresponding with its designated seismic category, as provided in PSAR table 3.7-2 and applied to the NIE seismic design in PSAR section 3.7 and (2) is located in the main electrical building (MEB) and other structures within the NI. The NRC staff's evaluation of seismic design basis is presented in section 3.7 of this SE. As stated in PSAR table 3.7-3, the NSRST portions of the NIE are categorized as SC-IIA for seismic safety significance to ensure they can perform the associated NSRST PSFs during and following a seismic event. The seismic design requirements associated with SC-IIA 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. PSAR section 7.3.8.1 states that the special treatment process, as discussed in PSAR section 4.2, will determine hazard event frequencies and any resulting design requirements necessary for the NIE to defend against applicable hazards. The NRC staff finds this reasonable given the preliminary NIE design.

Fire Protection (PDC 3)

The NRC staff reviewed PSAR section 7.3.8.1, which states that the NIE is protected by fire protection systems that are designed and tested in accordance with applicable NFPA standards and that the fire protection systems meet the requirements of 10 CFR 50.48(a) and PDC 3. PSAR section 8.6 provides a description of FPP including a list of program elements that are implemented in an integrated manner to meet PDC 3 to minimize the probability and effect of fires and explosions. The acceptability of the FPP is described in section 8.6 of this SE. Based on the above information, the NRC staff finds that the preliminary design of the NIE is consistent with PDC 3.

Environmental and Dynamic Effects Design Bases (PDC 4)

The NRC staff reviewed PSAR section 7.3.8.1, which states that the NIE is designed to accommodate the effects of and to be compatible with environmental conditions associated with normal operations, maintenance, testing, and AOOs, and that the electrical SSCs are protected from dynamic effects that may result from equipment failures and from external hazards by their locations, as discussed above in the evaluation for PDC 2. The locations for which the electrical SSCs and hazard protection strategies, processes, and methods will be considered at the OL application stage. This commitment satisfies PDC 4 requirements to appropriately protect the NIE system against dynamic effects, and the NRC staff finds that the preliminary design of the NIE is consistent with PDC 4.

Sharing of Structures, Systems, and Components (PDC 5)

The NRC staff reviewed PSAR section 7.3.8.1, which states that NSRST NIE SSCs are shared between reactor modules. Further, the PSAR states that the NIE implements independence of redundant trains, conforms to RG 1.75 on independence, and sharing will not significantly impair the ability of the NIE to support NSRST PSFs, including in the event of an AOO, DBE, or DBA in one reactor module and the orderly shutdown of the remaining modules. Since the NIE conforms to RG 1.75 on independence and has redundant trains, sharing of the NIE is acceptable and the preliminary design is consistent with PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

The NRC staff reviewed PSAR section 7.3.8.1, which states that the safety-significant NIE SSCs undergo periodic surveillances, online monitoring provides information to control room operators, functional testing is performed in accordance with the maintenance program, preoperational testing is performed as described in PSAR section 12.1.1.2, and startup testing is performed as described in PSAR section 12.1.1. As discussed in the safety significance subsection above, the NIE is a risk-significant SSC. 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 during the OL application stage based on final risk-significance determinations.

Since the NIE testing is performed in accordance with the maintenance program and online monitoring provides indications to control room operators, the NRC staff finds that the NIE is consistent with PDC 6.

Reactor Design (PDC 10)

The NRC staff reviewed PSAR section 7.3.1.8, which states that PDC 10 is a higher-level requirement that is met via application of PDC-CDC 17. The NIE is designed to ensure it performs the required PSFs, as discussed in PSAR section 7.3.8.1:

- PSF 1.1.3 - control reactivity with movable poisons
- PSF 1.2.2 - maintain core heat removal through active means
- PSF 1.2.2.2 - restore active heat removal
- PSF 1.3.2 - remove water/steam following isolation of water/steam sources
- PSF 2.2 - isolate HPB

The NRC staff reviewed the details of the NIE design provided in PSAR section 7.3.8. The acceptability of the elements to meet the PSFs applied to NIE are discussed in the PDC-CDC 17 description below. Therefore, PDC 10 requirements are satisfied by meeting the requirements in PDC-CDC 17. The evaluation for compliance of the NIE to PDC-CDC 17 is discussed later in this section.

Electric Power Systems (PDC 17)

The NRC staff reviewed PSAR table 1.4-2, which provides the applicable RGs, and PSAR table 7.3.8-2 provides a list of the codes and standards applicable to the NIE. RGs 1.93, 1.128, 1.212, and 1.75 support compliance with requirements of PDC 17. PSAR table 1.4-2 states that for RG 1.93, “the guidance is not applicable to the design and construction phase of the project and will be addressed in the application for the operating license.” Since the NIE is NSRST, the NRC staff finds that RG 1.93 is not applicable and can be used as good engineering practice. The NIE partially conforms to RG 1.128 and fully conforms to RG 1.212. The NRC staff finds the use of RG 1.128 and RG 1.212 acceptable for the NIE, related to the installation design and installation of the batteries and sizing of the batteries, respectively. PSAR table 1-4.2 further states that RG 1.129 will be addressed at the OL application stage. The NRC staff finds it acceptable that RG 1.129 is not applicable for the CP, that it is good engineering practice for the design, and applicable for the OL application since it addresses maintenance, testing, and replacement of the batteries. PSAR tables 1.4-2 and 1.4-2 state that the NIE conforms to RG

1.75, and the NRC staff finds that applying RG 1.75 is acceptable since it addresses independence for the NIE.

The NRC staff reviewed PSAR section 7.3.8.1, which states that passive design features of the plant do not rely on electrical power for performance of RSFs or to achieve safe shutdown and failure of the NIE cannot challenge RSFs. Since there are no SR electrical power sources, the NRC staff determined that the NIE preliminary design is consistent with PDC 17.

PSAR section 7.3.8.3 indicates that special treatment requirements selected for the NIE will be described in the FSAR and submitted as a part of a future OL application. Plant programs are evaluated in chapter 8 of this SE and the NRC staff recognizes that these programs provide preliminary insight into special treatments for the NIE. Specifically,

- PSAR section 8.1 describes the QAP and references TR, “Long Mott Energy, LLC Quality Assurance Program Description,” (hereafter referred to as LME QAPD) (ML24345A043). Chapter 19 of LME QAPD addresses QA for NSRST SSCs.

- PSAR section 8.3 describes the maintenance program to control the performance of maintenance activities in the plant. The NRC staff expects that the maintenance program will address, in part, NSRST systems.

- PSAR section 8.5 describes the environmental qualification program.

SE section 7.2 evaluates special treatments for NSRST SSCs. The NRC staff finds that the above special treatment for the NIE addresses QA, maintenance, testing, inspection, and environmental qualification, and are therefore, acceptable.

PSAR section 7.3.8.3 states that at the preliminary stage of design, no NIE component has been identified as incorporating FOAK technology, design, or configurations that would warrant special treatment testing. The NRC staff will evaluate the continued validity of the FOAK treatment at the OL application when the detailed design information for the NIE is available.

The NRC staff determined that the preliminary information provided in the PSAR, for the NIE, meets the requirements of 10 CFR 50.34(a), in part, because the PSAR provides the following information

- an overall description of the NIE system, including its functions, key design features, and performance and operation,

- a description of how the NIE supports the PSFs in PSAR section 7.3.8 subsections,

- a discussion of conformance of the NIE design with applicable PDC, and

- codes and standards applied to the NIE described in subsections within PSAR section 7.3.8

7.3.8.2 Conclusion

Based on the review described above, the NRC staff concludes that the information in PSAR section 7.3.8 is sufficient and meets the applicable guidance and regulatory requirements identified in this SE section and at the beginning of this chapter for the issuance of a CP. Specifically, the information provided:

- meets the requirements of 10 CFR 50.34(a)(3)(i) for the NIE by describing the preliminary design identifying the applicable PDC,

- meets the requirements of 10 CFR 50.34(a)(3)(ii) by describing how the NIE design satisfies the applicable PDC, and

- meets the requirements of 10 CFR 50.34(a)(3)(iii) by providing the general arrangement of the NIE, where available.

7.3.9 Nuclear Island Cooling Water System

PSAR section 7.3.9 describes the NICW which provides active, closed-loop cooling to certain safety-significant and non-safety-significant NI heat loads. The NICW comprises two subsystems: (1) RBCW and (2) helium services cooling water system (HSCW). The RBCW is classified as NSRST and provides active cooling to the RCSS control rod drive mechanism (CRDM) cooling jackets and HCS circulator cooling jackets. The HSCW is classified as NST and is described in PSAR section 1.1.4.

The preliminary RBCW design includes one common cooling loop with four redundant equipment trains and is shared between all four plant modules. Each equipment train includes a 50 percent capacity pump and a 50 percent capacity chiller such that any two chillers can meet the cooling water demands of all four modules simultaneously during normal operation. Additionally, each train is sized to simultaneously support two modules at full power and two modules at reduced power or in a normal shutdown configuration.

The RBCW pumps are located inside the NIAS and the RBCW chillers are located adjacent to the NIAS. The expansion and head tank is located on the roof of the SST. All system heat loads are located inside the SST.

The RBCW supports the following NSRST PSFs, which are required to demonstrate adequate DID during specific LBEs:

- PSF 1.1.3 - control reactivity with movable poisons
- PSF 1.2.2 - maintain core heat removal through active means

Interfaces with the following SSCs are needed for RBCW to support NSRST PSFs:

- Reactivity control and shutdown system (RCSS)
- Helium circulator system (HCS)
- Nuclear island electrical system (NIE)
- Distributed control system (DCS)

The RBCW does not support any SR RSFs.

In addition to PDCs 1-6, the following PDCs are applicable to the RBCW:

- PDC-CDC 13 - Instrumentation and control
- PDC-CDC 44 - Structural and equipment cooling

7.3.9.1 Technical Evaluation

Safety Significance

The RBCW is important to safety because it provides cooling water to the RCSS CRDM cooling jackets and HCS helium circulator cooling jackets. A loss of RBCW would cause a trip of all helium circulators of all reactors resulting in a four reactor pressurized loss of forced cooling as described in PSAR section 3.4.9. The RBCW is relied upon to demonstrate adequate DID to control reactivity with movable poisons and restore active heat removal through support of the RCCS and HCS, respectively, for the LBEs listed in PSAR table 5.5-3. The RBCW is designed to:

- actively reject heat from the RCSS thereby supporting the RCSS functions to support sufficient reactivity control to prevent the occurrence of reactor trips and circulator trips, as appropriate; and

- actively reject heat from the HCS, thereby supporting the HCS functions to restore/maintain primary flow and ensure that SARRDLs are not exceeded during AOOs.

The PSAR provides the NSRST PSFs supported by the RBCW to demonstrate adequate DID.

The PSAR identifies portions of the RBCW as NSRST with the overall system being categorized as NSRST. In section 5.5.1 of the SE, the NRC staff determined that the classification of RBCW system as NSRST system 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 7.3.9, which states that the safety-significant RBCW SSCs responsible for ensuring sufficiency of structural integrity, design, and construction are identified in PSAR tables 7.3.9-1 and 7.3.9-2. The RBCW is designed, fabricated, constructed, and tested in accordance with the QAP described in PSAR section 8.1. The NRC staff evaluates the QAP in section 8.1 of this SE. The PSAR states that the construction codes identified in table 7.3.9-1 provide comprehensive requirements for the materials, design, fabrication, installation, examination, and testing appropriate for the pressure-retaining components. These industry consensus codes address all structural failure modes with appropriate margins for the RBCW's low-temperature and low-pressure water design conditions. The RBCW SSCs will be designed in accordance with the approach identified in PSAR section 3.8, including the material qualification plan. The NRC staff's evaluation of PSAR section 3.8 is provided in SE section 3.8. As stated in PSAR table 7.3.9-1, some of the materials, codes of construction, and design

conditions have not yet been determined for RBCW components. If non-nuclear codes and standards are selected for components, justification will be provided with the OL application.

Based on the information provided above, the NRC staff determines that the RBCW is consistent with PDC 1.

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

The NRC staff reviewed PSAR section 7.3.9, which includes discussion of protection against natural phenomena for the RBCW. Many of the RBCW components are protected from natural phenomena by their location within the NIAS and SST, as described in PSAR section 3.7. Additionally, the PSAR states that the NIAS and SST provide protection from applicable natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions for NSRST SSCs located within the NIAS and SST. The expansion and head tank is located on top of the RB and is designed to withstand the DBHLs described in PSAR sections 6.1.1 and 6.1.3. The expansion and head tank is also designed such that adequate net positive suction head is available with the tank empty so that the RBCW remains capable of fulfilling its NSRST functions if the expansion and head tank were to become compromised. The other RBCW components located outside of structures will be designed to ensure that missiles or external flooding generated by applicable natural phenomena do not compromise their ability to perform any PSF.

The RBCW is designed to withstand seismic events and achieve performance levels corresponding with its designated seismic category, which is shown in PSAR table 3.7-2 and applied as described in PSAR section 3.7. The NRC staff's evaluation of the seismic design basis is presented in section 3.7 of the SE. The PSAR states that hazard protection strategies, processes, and methods are part of detailed design considerations that will incorporate insights from more detailed deterministic and probabilistic hazards assessments in the FSAR. The NRC staff finds this to be acceptable given that the preliminary information presented above shows reasonable assurance of protection against natural phenomena without loss of capability of HPS to perform its safety-significant functions.

Based on the information provided above, the NRC staff finds that the preliminary HPS design is consistent with PDC 2.

Fire Protection (PDC 3)

The NRC staff reviewed PSAR section 7.3.9.1, which includes the discussion that the design, location, and configuration of the RBCW minimize the probability and effects of fires and explosions and that the fire protection systems are designed, installed, and tested to protect the RBCW and other safety-significant SSCs. PSAR section 8.6 provides a description of FPP including a list of program elements that are implemented in an integrated manner to meet the requirements of 10 CFR 50.48(a) as well as PDC 3 to minimize the probability and effect of fires and explosions. These elements include, in part, building and system design and layout, including appropriate use of fire-resistant assemblies such as fire barriers, fire doors, fire dampers, fire penetration seals. PSAR section 8.6 also indicates that the FPP elements will be described in the operational FPP to be developed in conjunction with and described in the application for an OL. Moreover, the section states that the FPP will be designed in full conformance with the portions of RG 1.189 that are technically relevant to the design. The acceptability of the FPP is described in section 8.6 of this SE.

Based on the preliminary design features of the RBCW and the programmatic commitments described in PSAR section 8.6, the NRC staff finds that the fire protection measures for the RBCW are consistent with PDC 3 to minimize the probability and effects of fires and explosions.

Environmental and Dynamic Effects Design Basis (*PDC 4)*

The NRC staff reviewed PSAR section 7.3.9, which states that the NSRST RBCW SSCs are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, AOOs, DBEs, and DBAs commensurate with the safety significance of the functions to be performed. PSAR section 3.8 describes the structural evaluation methodology for NSRST metallic SSCs, including design, construction, material selection, environmental compatibility assessment, and the material qualification plan. The acceptability of the methodology is described in section 3.8 of this SE. The methods applied for the evaluation of dynamic effects from internal and external hazards and the determination of associated DBHLs are described in PSAR section 6.1.1. The acceptability of the methodology is described in section 6.1 of this SE. The PSAR states that hazard protection strategies, processes, and methods are part of detailed design considerations that will incorporate insights from more detailed deterministic and probabilistic hazards assessments in the FSAR. Hazards that may affect RBCW SSCs will be evaluated as the plant design is finalized and through the special treatment process. As such, appropriate design requirements will be assigned to ensure they can perform their PSFs. The NRC staff finds the commitment to perform such analysis at the FSAR stage to be reasonable given the preliminary and evolving nature of the RBCW design.

Based on the information provided, the NRC staff concludes that there is sufficient detail in the PSAR to demonstrate that the RBCW designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, AOOs, DBEs, and DBAs commensurate with the safety significance of the functions to be performed and are consistent with PDC 4.

Sharing of Structures, Systems, and Components (PDC 5)

The NRC staff reviewed PSAR section 7.3.9, which states that the RBCW is shared among all four units with four redundant equipment trains, each containing a 50 percent capacity pump and 50 percent capacity chiller. Any two pumps and any two chillers are designed to meet the cooling water needs of all four units simultaneously, and any single train is sized to provide cooling to two units at full power and two units at reduced power or shut down. Additionally, failure of the RBCW cannot challenge RSFs since SR SSCs that are actively cooled by the RBCW are designed to fail safe upon a loss of cooling and passive design features do not rely on active cooling for the performance of RSFs. The NRC staff finds that the preliminary RBCW design is consistent with PDC 5 due to it being sized to perform the safety functions for all four modules simultaneously and having multiple equipment redundancies.

Monitoring, Inspection, Testing, and Surveillance (*PDC* 6)

The NRC staff reviewed PSAR section 7.3.9, which includes the discussion of the application of PDC 6 to the RBCW and states that safety-significant RBCW SSCs are designed to permit appropriate monitoring, periodic inspection, testing, and/or surveillances to ensure functional capability commensurate with the safety-significant functions to be performed. The design provides for functional testing with conditions as close to design as practical to ensure the

functionality and performance of RBCW safety-significant components. Passive NSRST RBCW SSCs are included in the RAM program, which is described in PSAR section 8.10 and evaluated in section 8.10 of the SE. The RBCW design provides access for implementing remote preservice and inservice requirements such as online monitoring. Active portions of the RBCW are tested for operability and performance as part of the maintenance program, which is described in PSAR section 8.3 and evaluated in section 8.3 of the SE. PSAR section 7.3.9 stated the potential IST requirements include exercise testing and diagnostic testing. The planned functional testing is expected to encompass the conditions and ranges associated with operation of the RBCW; 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 inservice inspection and IST programs at the OL application stage.

Instrumentation and Control (*PDC-CDC 13)*

The NRC staff reviewed PSAR section 7.3.9, which states that instrumentation is provided in appropriate locations within the RBCW to monitor and control flow, pressure, and temperature over their anticipated ranges during normal operations and certain LBEs to support and provide the indication of functions performed by safety-significant SSCs and maintain those variables within prescribed operating ranges. The DCS provides controls for the RBCW equipment needed to support NSRST PSFs. The NRC staff finds that the preliminary design of the RBCW is consistent with PDC-CDC 13 because instrumentation and an interface with DCS is provided to monitor and control RBCW parameters within prescribed operating ranges and support the performance of NSRST PSFs.

Structural and Equipment Cooling (*PDC-CDC 44)*

The NRC staff reviewed PSAR section 7.3.9.1, which states that the RBCW is designed to provide an active means of heat transfer from the safety-significant RCSS and HCS to a heat sink to ensure that SARRDLs are not exceeded during AOOs. The RBCW removes heat from the RCSS CRDMs and the HCS helium circulator motors by providing water to the cooling jackets of these components. The heat is ultimately rejected to the atmosphere by chillers to maintain cooling water temperature within acceptable limits. Based on the review performed, the NRC staff finds that the preliminary RBCW design is consistent with PDC-CDC 44 because of the system's ability to remove heat from the safety-significant RCSS CRDMs and HCS motors.

Safety Functions

PSAR section 7.3.9.1 states that the RBCW supports the following NSRST PSFs and PSF CDCs:

- PSF 1.1.3 - control reactivity with movable poisons
    - CDC for PSF 1.1.3.1 - The RBCW is designed to actively reject heat from the RCSS, thereby supporting the RCSS functions to support sufficient reactivity control to prevent the occurrence of reactor trips and circulator trips, as appropriate, during AOOs.

- PSF 1.2.2 - maintain core heat removal though active means

    - CDCs for PSFs 1.2.2.2.1, 1.2.2.2.2, 1.2.2.2.3, and 1.2.2.2.4 - The RBCW is designed to actively reject heat from the HCS, thereby supporting the HCS functions, to restore/maintain primary flow and ensure SARRDL are not exceeded during AOOs.

The preliminary design of the RBCW supports these PSFs through its ability to actively remove heat from the RCSS CRDMs and the HCS helium circulators as discussed above.

Special Treatments

PSAR section 7.3.9 states that final special treatments for the RBCW will be identified and selected following expansion of the design and PRA to align with the development of system-level reliability and capability targets, consistent with the process described in NEI 18-04, as endorsed by RG 1.233. A general description of special treatments applied to NSRST SSCs is described in PSAR section 7.2.

The NRC staff finds that while special treatments for RBCW components in general have not yet been identified, the preliminary special treatments are provided through compliance with PDC 6. Additionally, preliminary special treatments under consideration for the RBCW include 100 percent volumetric examination (UT or RT) of pressure boundary fusion welds and requiring full penetration pressure boundary fusion welds.

The programs described in PSAR chapter 8 and chapter 12 are preliminarily identified in the PSAR as programmatic special treatments to be applied to safety-significant SSCs.

Based on its review of the functions of the RBCW against the description of the codes and standards and qualification programs described in the PSAR, the NRC staff determined that the preliminary special treatments description and commitments are appropriate for the PSAR stage. A final determination of sufficiency of special treatments applied to the RBCW will be performed at the OL application stage.

PSAR section 7.3.9.3 states that at the preliminary stage of design, no RBCW component has been identified as incorporating FOAK technology, design, or configurations that would warrant special treatment testing. The NRC staff will evaluate the continued validity of the FOAK treatment at the OL application when the detailed design information for the RBCW is available.

7.3.9.2 Conclusion

Based on its review, the NRC staff concludes that the information provided in PSAR section 7.3.9 is sufficient and meets the applicable guidance and regulatory requirements identified above for the issuance of a CP. Specifically, the information provided:

- meets the requirements of 10 CFR 50.34(a)(3)(i) for the RBCW by describing the preliminary design identifying the applicable PDC,

- meets the requirements of 10 CFR 50.34(a)(3)(ii) by describing how the RBCW design satisfies the applicable PDC, and

- meets the requirements of 10 CFR 50.34(a)(3)(iii) by providing the materials of construction and general arrangement of the RBCW, where available.

7.3.10 Nuclear Island Heating, Ventilation, and Air Conditioning

PSAR section 7.3.10 describes the NIHV. The NIHV provides conditioned air to rooms that house SR, NSRST, and NST equipment through subsystems that separately cool individual structures.

The PSAR states that the NIHV comprises the following subsystems:

- NI module HVAC system (MHV)
- Helium service facility HVAC system (HSHV)
- Radwaste building HVAC system (RWHV)
- Reactor building HVAC system (RBHV)
- Inter-unit access tunnel HVAC system (CEHV)
- Canister processing facility HVAC system (CPHV)
- Local MEB HVAC supplementary system
- Local CEB HVAC supplementary system
- Local NIAS HVAC supplementary system
- FHAB supplementary cooling system

The NI HVAC system consists of a primary loop that conditions the access and security building, HSF, FHAB, CPF, IUAT, SST, RWB, and local subsystems that provide HVAC services to the MEB, CEB, and NIAS. In addition to the local subsystems, the FHAB is equipped with a local supplementary cooling system to provide the necessary cooling to rooms that house SR equipment.

The NIHV system preliminary design provides equipment cooling to NIE that supports the following NSRST PSFs:

- PSF 1.1.3 - controlling reactivity with movable poisons;
- PSF 1.2.2 - maintaining core heat removal through active means;
- PSF 1.3.2 - removing water/steam from the HPB following isolation of water/steam sources; and
- PSF 2.2 - isolating the HPB.

The NIHV system does not support any SR RSFs, which rely on passive design features of the facility.

In addition to PDC 1-6, the following PDCs are applicable to the NIHV:

- PDC-CDC 13 - Instrumentation and control
- PDC-CDC 44 - Structural and equipment cooling

7.3.10.1 Technical Evaluation

Safety Significance

The NIHV system is important to safety because it provides equipment cooling to NIE components that supply required electrical power to plant SSCs during certain LBEs to perform NSRST PSFs. A loss of NIHV will result in a loss of cooling to NIE components that supply required electrical power to plant SSCs, which will eventually cause those SSCs to degrade or fail. Loss of cooling to SR SSCs cooled by the NIHV will not result in a loss of the SR SSC to perform its RSFs. The NIHV is also relied upon to demonstrate adequate DID in supporting several NSRST PSFs for the LBEs listed in PSAR table 5.5-6. The NIHV system is designed to:

- Provide cooling to areas containing NIE components that provide electrical power to the NIS, IPS, RCSS, and RBCW, thereby supporting their respective functions that support NSRST PSFs to support sufficient reactivity control to prevent the occurrence of reactor trips and circulator trips, as appropriate.

- Provide cooling to areas containing NIE components that provide electric power to the HCS, SSS, NIS, DCS, and RBCW, thereby supporting their respective functions that support NSRST PSFs to restore/maintain active heat removal and ensure that SARRDLs are not exceeded during AOOs.

- Provide cooling to areas containing NIE components that provide electric power to the RCCS, thereby supporting the RCCS functions that support NSRST PSFs to control heat removal from the reactor cavity through active means and ensure SARRDLs are not exceeded during AOOs.

- Provide cooling to areas containing NIE components that provide electric power to the SGS, SSS, NIS, and IPS, thereby supporting their respective functions that support NSRST PSFs to control water flow into the SG, dump SG inventory, and ensure that SARRDLs are not exceeded during AOOs.

- Provide cooling to areas containing NIE components that provide electric power to the FHS, SGS, NIS, and IPS, thereby supporting their respective functions that support NSRST PSFs to isolate the HPB during applicable depressurization LBEs.

The PSAR provides the NSRST PSFs supported by the NIHV system to demonstrate adequate DID.

The PSAR identifies portions of the NIHV system as NSRST with the overall system being categorized as NSRST. NST portions of the NIHV system are not required to support NSRST PSFs. In section 5.5.1 of the SE, the NRC staff determined that the classification of NIHV system as NSRST 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 7.3.10, including table 7.3.10-1, which lists HVAC system codes and standards that address ventilation system performance related to temperature, air quality, and smoke removal functions, among others. Records of the design, fabrication, construction, and testing are maintained in accordance with the QAP described in PSAR section 8.1. The NRC staff evaluates the QAP in section 8.1 of this SE. The PSAR states that ASME AG-1 may be applied to NIHV subsystems as a special treatment. Based on the information provided above and the limited safety significance of the NIHV, the NRC staff determines that the NIHV is consistent with PDC 1.

Design Bases for Protection Against Natural Phenomena (PDC 2)

The NRC staff reviewed PSAR section 7.3.10, which includes discussion of protection of natural phenomena for the NIHV system. Many of the NIHV components are protected from natural phenomena by their location within the RB, NIAS, FHAB, MEB, and CEB. As discussed in PSAR section 3.7 and chapter 7, these structures are designed to provide protection from applicable natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions. The PSAR states that NSRST NIHV components that are located outside will be designed to ensure that missiles or external flooding generated by natural phenomena does not compromise their ability to perform any PSF. If the special treatment process determines that these SSCs require external flooding or missile protection, the final design will incorporate appropriate missile barriers or other protective features to ensure NIHV components perform their PSFs.

As stated in PSAR table 3.7-2, the NSRST portions of the NIHV are categorized as SC-IIA for seismic safety significance to ensure they can perform the associated NSRST PSFs during and following a seismic event. The seismic design requirements associated with SC-IIA 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. The PSAR states that hazard protection strategies, processes, and methods are part of detailed design considerations that will incorporate insights from more detailed deterministic and probabilistic hazards assessments in the FSAR. The NRC staff finds this to be acceptable given the preliminary information presented above shows reasonable assurance of protection against natural phenomena without loss of capability of HPS to perform its safety-significant functions.

Based on the information provided above, the NRC staff determines that the preliminary design of the NIHV is consistent with PDC 2.

Fire Protection (PDC 3)

The NRC staff reviewed PSAR section 7.3.10, which includes discussion that the design, location, and configuration of the NIHV system minimize the probability and effects of fires and explosions and that the fire protection systems are designed, installed, and tested to protect the NIHV system and other safety-significant SSCs. PSAR section 8.6 provides a description of the FPP including a list of program elements that are implemented in an integrated manner to meet the requirements of 10 CFR 50.48(a) as well as PDC 3 to minimize the probability and effect of

fires and explosions. These elements include, in part, building and system design and layout, including appropriate use of fire-resistant assemblies such as fire barriers, fire doors, fire dampers, fire penetration seals. PSAR section 8.6 also indicates that the FPP elements will be described in the operational FPP to be developed in conjunction with and described in the application for an OL. Moreover, the section states that the FPP will be designed in full conformance with the portions of RG 1.189 that are technically relevant to the design. The acceptability of the FPP is described in section 8.6 of this SE.

Based on the preliminary design features of the NIHV system and the programmatic commitments described in PSAR section 8.6, the NRC staff finds that the fire protection measures for the NIHV are consistent with PDC 3 to minimize the probability and effects of fires and explosions.

Environmental and Dynamic Effects Design Basis (PDC 4)

The NRC staff reviewed PSAR section 7.3.10, which states that the NSRST SSCs are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, AOOs, DBEs, and DBAs. The PSAR also states that the methods applied for the evaluation of dynamic effects from hazards and the determination of DBHLs are described in PSAR section 6.1.1. The acceptability of the methodology is described in section 6.1 of this SE. The PSAR states that hazards protection strategies, processes, and methods are part of detailed design considerations that will incorporate insights from more detailed deterministic and probabilistic hazards assessments in the FSAR. Hazards that may affect NIHV SSCs will be evaluated as the plant design is finalized and through the special treatment process. As such, appropriate design requirements will be assigned to ensure they can perform their PSFs. The NRC staff finds the commitment to perform such analysis at the FSAR stage to be reasonable given the preliminary and evolving nature of the NIHV design.

Based on the information provided, the NRC staff concludes that there is sufficient detail in the PSAR to demonstrate that the NIHV is designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, AOOs, DBEs, and DBAs commensurate with the safety significance of the functions to be performed and is consistent with PDC 4. Considering the limited safety-significant function of the NIHV system of providing equipment cooling, the NRC staff finds the level of detail provided to be acceptable for the CP review.

Sharing of Structures, Systems, and Components (PDC 5)

The NRC staff reviewed PSAR section 7.3.10, which states that the NIHV is shared among all four units with five 25 percent supply air handler units (AHUs), four 50 percent capacity energy recovery units, and four 50 percent capacity exhaust AHUs. Additionally, there is a supplementary cooling system provided to the supply the AHUs needed to maintain NSRST NIHV functionality in the event that the HSCW is unable to provide sufficient cooling. In addition to the backup water-side cooling provided to the AHUs, the supplementary cooling system consists of backup direct expansion units for the MEB and NIAS electrical rooms. Additionally, failure of the NIHV cannot challenge RSFs since SR SSCs that are actively cooled by the NIHV are designed to fail safe upon a loss of cooling and passive design features do not rely on active cooling for performance of RSFs. The NRC staff finds that the preliminary NIHV system design

is consistent with PDC 5 due to it being sized to perform the safety functions for all four units simultaneously and having multiple equipment redundancies.

Monitoring, Inspection, Testing, and Surveillance (PDC 6)

The NRC staff reviewed PSAR section 7.3.10, which states that periodic monitoring of the passive components is performed in accordance with the maintenance program, which is evaluated in SE section 8.3. The design provides access for implementing remote preservice and inservice requirements. The NRC staff reviewed the identified potential MANDE requirements, including visual examination and finds the MANDE requirements appropriate to provide adequate assurance to maintain the functional capability of the NIHV. Active portions of the NIHV are tested for operability and performance as part of the maintenance program which is described in PSAR section 8.3 and evaluated in section 8.3 of the SE. PSAR section 7.3.10 stated that the potential IST requirements include exercise testing and diagnostic testing. However, the maintenance program does not specify a method for developing and implementing an inspection and testing program and the NRC staff will review the final maintenance program at the OL application. In addition, LME states that the preoperational testing, as described in PSAR section 12.1.1 and evaluated in SE section 12.1, will be performed to provide a baseline for the inservice inspections.

Instrumentation and Control (PDC-CDC 13)

The NRC staff reviewed PSAR section 7.3.10, which states that instrumentation is designed to monitor NIHV differential pressures, flow, exhaust radiation, SSC status, and other variables over their anticipated ranges during normal operations and LBEs, as appropriate, to provide indication of functions performed by safety-significant SSCs. Additionally, NIHV parameters are monitored and controlled by the DCS to maintain those variables within prescribed operating ranges. The NRC staff finds that the preliminary design is consistent with PDC-CDC 13 based on the information discussed above.

Structural and Equipment Cooling (PDC-CDC – 44)

The NRC staff reviewed PSAR section 7.3.10, which states that the NIHV is designed to provide cooling to areas containing NIE components that provide electrical power supply and electrical protection for the following safety-significant systems: RCSS, HCS, SGS, RCCS, SSS, FHS, NIS, IPS, DCS, NICW, and NIHV. PSAR section 6.4.13.2 states that the NIHV provides cooling to RPS equipment but is not relied upon for RPS to perform its RSFs. The MEB, CEB, and electrical rooms in the NIAS are cooled by local wall-mounted packaged units. The MEB also has a series of louvers and ventilation fans to cool the diesel generator room of the MEB. The FHAB supplementary cooling system is designed to remove heat from electrical rooms with equipment that performs SR functions during a loss of power event. PSAR section 7.3.10.2 states that during a loss of power event, FHAB supplementary cooling, MEB cooling, MEB ventilation, and CEB cooling are placed on backup power to remain operational. These subsystems and components provide necessary heat removal to NIE equipment needed to support NSRST PSFs. Based on the review performed, the NRC staff finds that the preliminary NIHV design is consistent with PDC-CDC 44 because of its capability to actively remove heat from the safety-significant SSCs discussed above.

Safety Functions

PSAR section 7.3.10 states that the NIHV supports the following PSFs and PSF CDCs:

- PSF 1.1.3 - control reactivity with movable poisons
    - CDC for PSF 1.1.3.1 – The NIHV is designed to provide to areas containing NIE components that provide electric power to the NIS, IPS, RCSS, and RBCW, thereby supporting their respective functions that support NSRST PSFs, described in PSAR sections 6.4.13, 7.3.11, 6.4.10, and 7.3.9, respectively, to support sufficient reactivity control to prevent the occurrence of reactor trips and circulator trips, as appropriate, during certain LBEs.
- PSF 1.2.2 - maintain core heat removal though active means
    - CDCs for PSFs 1.2.2.2.1, 1.2.2.2.2, 1.2.2.2.3, and 1.2.2.2.4 - The NIHV is designed to provide cooling to areas containing NIE components that provide electric power to the HCS, SSS, NIS, DCS, and RBCW, thereby supporting their respective functions that support NSRST PSFs, described in PSAR sections 7.3.5, 7.3.4, 6.4.14, 7.3.12 and 7.3.9, respectively, to restore/maintain active heat removal and ensure that SARRDL are not exceeded during AOOs.
    - CDC for PSF 1.2.2.3 - The NIHV is designed to provide cooling to areas containing NIE components that provide electric power to the RCCS, thereby supporting the RCCS functions that support NSRST PSFs, described in PSAR section 6.4.11, to control heat removal from the reactor cavity through active means and ensure SARRDL are not exceeded during AOOs.
- PSF 1.3.2 - remove water/steam following isolation of water/steam sources
    - CDC for PSF 1.3.2.1 - The NIHV is designed to provide cooling to areas containing NIE components that provide electric power to the SGS, SSS, NIS and IPS, thereby supporting their respective functions that support NSRST PSFs, described in PSAR sections 6.4.12, 7.3.4, 6.4.14, and 7.3.11, respectively, to control water flow into the SG, dump SG inventory, and ensure that SARRDL are not exceeded during AOOs.
- PSF 2.2 - isolate HPB
    - CDC for PSF 2.2.1 - The NIHV is designed to provide cooling to areas containing NIE components that provide electric power to the FHS, SGS, NIS and IPS, thereby supporting their respective functions that support NSRST PSFs, described in PSAR sections 7.3.6, 6.4.12, 6.4.14, and 7.3.11, respectively, to isolate the HPB during depressurization certain LBEs.

The preliminary design of the NIHV supports these PSFs by actively removing heat from the areas containing NIE components that provide electrical power supply to and electrical protection for the RCSS, HCS, SGS, RCCS, SSS, FHS, NIS, IPS, DCS, RBCW, and NIHV during AOOs and certain additional LBEs.

Special Treatments

PSAR section 7.3.10 states that final special treatments for the NIHV will be identified and selected following the expansion of the design and PRA to align with the development of system-level reliability and capability targets, consistent with the process described in NEI 18-04, as endorsed by RG 1.233. A general description of the special treatments applied to NSRST SSCs is described in PSAR section 7.2.

The NRC staff finds that while special treatments for NIHV components in general have not yet been identified, the preliminary special treatments are provided through compliance with PDC 6. Additional special treatment being considered is the application of ASME AG-1.

The programs described in PSAR chapter 8 and chapter 12 are preliminarily identified in the PSAR as programmatic special treatments to be applied to safety-significant SSCs.

Based on its review of the functions of the NIHV system against the description of the codes and standards and qualification programs described in the PSAR, the NRC staff determined that the preliminary special treatments description and commitments are appropriate for the PSAR stage. A final determination of sufficiency of special treatments applied to the NIHV system will be performed at OL application stage.

PSAR section 7.3.10.3 states that at the preliminary stage of design, no NIHV component has been identified as incorporating FOAK technology, design, or configurations that would warrant special treatment testing. The NRC staff will evaluate the continued validity of the FOAK treatment at the OL application when the detailed design information for the NIHV is available.

7.3.10.2 Conclusion

Based on its review, the NRC staff concludes that the information provided in PSAR section 7.3.10 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and in the beginning of this chapter for the issuance of a CP. Specifically, the information provided:

- meets the requirements of 10 CFR 50.34(a)(3)(i) for the NIHV by describing the preliminary design identifying the applicable PDC,
- meets the requirements of 10 CFR 50.34(a)(3)(ii) by describing how the NIHV design satisfies the applicable PDC, and
- meets the requirements of 10 CFR 50.34(a)(3)(iii) by providing the materials of construction and general arrangement of the NIHV, where available.

7.3.11 Investment Protection System

PSAR section 7.3.11 describes the IPS. The IPS is a subsystem of the PCDAS. The IPS is a digital I&C system that monitors parameters and provides automatic protective actions for variables essential to responding to AOOs and preventing damage to key systems and components. The IPS accomplishes this by detecting when parameters deviate from allowable

normal operating ranges and initiating protective actions that force mode or state transitions of supported systems.

The IPS provides DID by supporting NSRST PSFs to minimize the frequency of challenges to the RPS. The IPS only initiates protective actions if the DCS fails to maintain the selected plant parameter within acceptable limits. The IPS accepts input signals from field instrumentation, processes logic, and generates output signals to automatically initiate the required combination of the following protective actions:

- Reactor shutdown via the RCSS
- Helium circulator system (HCS) rundown
- Steam generator system (SGS) isolation
- SGS inventory dump
- Fuel handling system (FHS) isolation
- Helium service system (HSS) isolation

The IPS utilizes override features to prioritize IPS protective actions over DCS controls. IPS components are installed in the FHAB, the NIAS, and the MEB and are unit specific. No operator actions or operator controls or displays are needed for the IPS to support NSRST PSFs.

The system-specific regulatory requirements for the IPS are as follows:

- 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).

In addition to PDCs 1-6, the following PDCs are applicable to the IPS:

- PDC 10 - Reactor design
- PDC-CDC 13 - Instrumentation and control
- PDC 20 - Protection system function
- PDC 24 - Separation of protection and control systems
- PDC 25 - Protection system requirements for reactivity control malfunctions
- PDC-CDC 26 - Reactivity control systems
- PDC-CDC 30 - Integrity of reactor helium pressure boundary

7.3.11.1 Technical Evaluation

Safety Significance

PSAR section 7.3.11 states that the IPS is a subsystem of the PCDAS and is classified as NSRST. The IPS provides DID by performing three NSRST PSFs to minimize the frequency of challenges to the RPS. PSAR table 5.5-2 lists the PSFs and the specific LBEs that the IPS is designed to prevent or mitigate for DID adequacy. Specifically, the IPS provides actuation

signals to the RCSS and HCS to support reactivity control. The IPS also provides actuation signals to the SGS to cease water flow from the SG to preclude primary system pressure relief, remove water and steam from the SG, and ensure that SARRDLs are not exceeded during AOOs. Finally, the IPS provides actuation signals to the FHS and the SGS to isolate the HPB during certain depressurization LBEs.

As confirmed by the NRC staff via RCI (ML26239A157), LME is in the process of incorporating the results of updated risk-significance evaluations following the NEI 18-04 methodology. The preliminary results of these evaluations indicate that the IPS is risk significant, which would mean that failure of the prevention or mitigation function of the IPS could lead some of the LBEs crediting IPS to exceed the F-C target.

The PSAR identifies portions of the IPS as NSRST with the overall system being categorized as NSRST. In section 5.5.1 of the SE, the NRC staff determined that the classification of IPS as an NSRST system is consistent with the guidance in NEI 18-04, as endorsed with clarifications by RG 1.233. Based on the information provided in the PSAR, the NRC staff finds that the IPS classification as NSRST is reasonable.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR table 7.3.11-1, which identifies the preliminary consensus codes and standards applicable to the IPS. These codes and standards address areas including quality and equipment qualification. Specifically, International Electrotechnical Commission (IEC) 61508, “Functional Safety of Electrical/Electronic/Programmable Electronic Safety-Related Systems,” is listed as one of the applicable standards. While IEC 61508 is a non-nuclear specific standard and has not been reviewed or endorsed by the NRC for use in SR I&C, it is an internationally recognized, risk-informed, performance-based standard developed for industrial safety applications. Given the NSRST classification, use of IEC 61508 is reasonable for the IPS. Additionally, the IPS is designed, fabricated, constructed, and tested in accordance with the QAP described in part III of the LMGS QAPD and PSAR section 8.1. Additional special treatments related to QA 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 QA requirements under PDC 1 for the NSRST portions of the IPS. 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 NRC staff’s review of the information provided in the PSAR regarding the codes and standards applicable to the IPS and the evaluation in section 8.1, the NRC staff finds that the preliminary design of the IPS is consistent with PDC 1.

Design Basis for Protection Against Natural Phenomena (PDC 2)

The NRC staff reviewed PSAR section 7.3.11.1 which states that the IPS is designed to withstand the effects of natural phenomena and the determination of associated DBHLs are described in PSAR sections 6.1.1. and 6.1.3 and evaluated in section 6.1 of this SE. DBHL related requirements are applied to the NSRST IPS to prevent interactions with SR SSCs that would impact the ability of the SSC to perform an RSF. As stated in PSAR table 3.7-3, the NSRST portion of the IPS is categorized as SC-IIA for seismic safety significance to ensure the NSRST portion of the IPS can perform the associated NSRST PSFs during and following a seismic event. The seismic design requirements associated with SC-IIA and the codes and

standards applied are described in PSAR section 3.7 and evaluated in section 3.7 of this SE. The special treatment process, described in PSAR section 4.2 and evaluated in section 4.2 of this SE, determines the specific hazard event frequencies and any resulting design requirements necessary for the IPS. Based on the NRC staff's review of information provided in the PSAR on how the IPS is designed to withstand the effects of natural phenomena and the evaluations in sections 3.7, 4.2, and 6.1, the NRC staff finds that the preliminary design of the IPS is consistent with PDC 2.

Fire Protection (PDC 3)

The NRC staff reviewed PSAR section 7.3.11.1, which states that the IPS is designed, located, and configured to minimize the probability and effect of fires and explosions through the selection of noncombustible and non-fire sustaining materials, to the extent practical. Further, the IPS is protected by fire protection systems designed, installed, and tested in accordance with applicable NFPA standards and as determined by the FHA and FSSA to be necessary to protect safety-significant SSCs and to ensure performance of safe-shutdown functions. These systems are a component of the comprehensive plant FPP and are designed to meet the requirements of 10 CFR 50.48(a)(1) and are evaluated in section 8.6 of this SE. Based on the NRC staff's review of information provided in the PSAR on the fire protection features designed to meet the requirements of 10 CFR 50.48(a)(1) and the evaluation in section 8.6 of this SE, the NRC staff finds that the preliminary design of the IPS is consistent with PDC 3.

Environmental and Dynamic Effects Design Bases (PDC 4)

The NRC staff reviewed PSAR section 7.3.11.1, which states that the IPS 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 IPS components are separated by train as necessary and protected from dynamic effects that may result from equipment failures and from external hazards. As stated in PSAR section 6.1.3 and evaluated by the NRC staff in section 6.1 of this SE, DBHL related requirements are applied to the NSRST IPS to prevent interactions with SR SSCs that would impact the ability of the SSC to perform an RSF. Based on the NRC staff's review of information provided in the PSAR on the IPS design consideration of environmental conditions and effects and the evaluation in section 6.1, the NRC staff finds that the preliminary design of the IPS is consistent with PDC 4.

Sharing of Structures, Systems, and Components (PDC 5)

The NRC staff reviewed PSAR section 7.3.11.1, which states that each reactor unit features a dedicated IPS. Based on the NRC staff's review, the NRC staff finds that the preliminary design of the IPS is consistent with PDC 5 since the IPS is not shared between reactor units.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

As discussed in the safety significance subsection above, LME is in the process of incorporating the results of updated risk-significance evaluations following the NEI 18-04 methodology. The preliminary results of these evaluations indicate that the IPS is risk significant. 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 during the OL application based on final risk-significance determinations. The NRC staff reviewed PSAR section 7.3.11.1, which states that the safety-significant IPS SSCs are designed with indications

from online monitoring techniques for operators. The IPS also meets applicable portions of PDC 6 through a combination of preservice and inservice requirements. Safety-significant IPS SSCs undergo periodic surveillances. Additional preoperational and startup testing is described in PSAR sections 12.1.1.2 and 12.1.1 and evaluated in section 12.1 of this SE. Baseline testing along with monitoring, testing, and surveillance requirements provide confidence that the capability of the IPS is retained through plant life. Based on the NRC staff's review of the information provided in the PSAR on the RIM and IST programs and the NRC staff's evaluation in sections 8.2 and 8.4 of this SE, online monitoring capabilities, application of preservice and inservice requirements, and the testing requirements evaluated in section 12.1, the NRC staff finds that the preliminary design of the IPS is consistent with PDC 6.

Reactor Design (PDC 10)

The NRC staff reviewed section 7.3.11.1, which states that PDC 10 is a higher-level requirement that is met in the IPS design via application of PDC 20, PDC 25, and PDC-CDC 26. The NRC staff finds that this statement is appropriate because it is consistent with the NRC-approved Xe-100 PDC TR. As discussed in the basis for PDC 10 in the Xe-100 PDC TR, PDC 10 is a design limit that supports acceptance criteria in PDC 20, PDC 25, and PDC-CDC 26. As discussed in the basis for PDC 10 in the Xe-100 PDC TR, PDC 10 is a design limit that supports PDC. Therefore, PDC 10 requirements are satisfied by meeting the requirements in PDC 20, PDC 25, and PDC-CDC 26. The evaluations for compliance of the IPS design to these PDC are performed later in this section.

Instrumentation and Control (PDC-CDC 13)

The NRC staff reviewed PSAR section 7.3.11.1, which states that the IPS 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 PSAR also states that the IPS monitors plant parameters by collecting information from NIS field sensors and initiates automatic protective actions when monitored parameters reach or exceed specified setpoints, prior to reaching or exceeding RPS setpoints. Based on the NRC staff's review, the NRC staff finds the preliminary design of the IPS to have I&C necessary to accomplish PSFs and to support overall DID consistent with PDC-CDC 13.

Protection System Functions (PDC 20)

The NRC staff reviewed PSAR section 7.3.11.1, which states that the IPS 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, and to support NSRST PSFs when conditions requiring initiation are sensed. This is accomplished using I&C designed to meet PDC-CDC 13 as discussed in this section. 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 IPS is designed to perform these functions by utilizing setpoints which are selected such that the IPS responds prior to plant parameters reaching or exceeding RPS setpoints and provides DID. The RPS is designed to respond if an AOO is unmitigated by the IPS, as described in PSAR section 6.4.13 and evaluated in section 6.4.13 of this SE. Based on the NRC staff's review of the information provided in the PSAR on the IPS PSFs and the functional independence of the RPS evaluated

in section 6.4.13, the NRC staff finds that the preliminary design of the IPS is consistent with PDC 20.

Separation of Protection and Control Systems (PDC 24)

The NRC staff reviewed PSAR section 7.3.11.1, which states that the IPS is functionally independent from the RPS and from the DCS. The IPS and DCS share a standard communications bus such that all IPS sensor information is also available to the DCS; however, the IPS communication bus has a higher priority than the DCS communication bus. To maintain independence from the DCS, all IPS input signals needed for support of NSRST PSFs are directly connected to the IPS and all IPS output signals are directly connected to field actuation devices. The field actuation devices utilized by the IPS are diverse from those utilized by the RPS for similar protective actions. As documented in section 6.4.13 of this SE, the NRC staff determined that the RPS is designed with sufficient separation from the control systems and is therefore consistent with PDC 24. Based on the NRC staff's review of the independence and diverse features of the IPS and the evaluation in section 6.4.13, the NRC staff finds that the preliminary design of the IPS is consistent with PDC 24.

Protection System Requirements for Reactivity Control Malfunctions (PDC 25)

The NRC staff reviewed PSAR section 7.3.11.1, which states that the IPS is designed to ensure that SARRDLs are not exceeded during AOOs, accounting for a single malfunction of the reactivity control systems. The NRC staff confirmed that the IPS is designed to perform this DID function by initiating the appropriate combination of protective actions and if unmitigated by the IPS, the functionally independent RPS is designed to respond accordingly. The RPS design is evaluated in section 6.4.13 of this SE and is determined to be consistent with PDC 25. Based on the NRC staff's review of the information provided in the PSAR and the evaluation in section 6.4.13, the NRC staff finds that the preliminary design of the IPS is consistent with PDC 25.

Reactivity Control Systems (PDC-CDC- 26)

The NRC staff reviewed PSAR section 7.3.11.1, which states that the IPS is designed to provide a means, which is independent and diverse from the reactivity control RFDC, to insert negative reactivity at a sufficient rate and amount to ensure, with appropriate margin for malfunctions, that SARRDLs and HPB design limits are not exceeded, and safe shutdown is achieved and maintained during certain LBEs. The RCSS is described in PSAR section 6.4.10 and consists of rod banks capable of shutting down the reactor and maintaining safe shutdown. The IPS is designed to perform this function by initiating a reactor trip function by inserting negative reactivity by controlled neutron absorbing rod (NAR) stepwise movements at a sufficient rate and amount to ensure safe shutdown is achieved and maintained during certain LBEs. The NRC staff determined that the IPS function is diverse and independent from the RPS, which 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 evaluation of the information provided in the PSAR on the ability of the IPS to insert negative reactivity via the RCSS to accomplish PSFs, the NRC staff finds that the preliminary design of the IPS is consistent with PDC-CDC 26.

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

PSAR section 7.3.11.1 states that the IPS is designed to provide a means to detect and isolate sources of helium leakage during certain LBEs. The NRC staff determined that the IPS is

designed to perform these functions by initiating an HPB isolation via interface with the FHS and SGS. The isolation function is supported by the IPS I&C designed to monitor variables over their anticipated ranges, as evaluated above to meet PDC-CDC 13. Based on the NRC staff's review of the isolation capabilities of the IPS via interface with the FHS and SGS to maintain the integrity of the HPB to accomplish PSFs, the NRC staff finds that the preliminary design of the IPS is consistent with PDC-CDC 30.

Safety Functions

PSAR section 7.3.11.1 states that the IPS is designed to support the NSRST PSF 1.1.3, control reactivity with movable poisons; PSF 1.3.2, remove water/steam following isolation of water/steam sources; PSF 2.2, isolate HPB; and PSF 2.2.1, isolate systems connected to HPB.

LME identified PDC-CDC 13, PDC-CDC 26, and PDC-CDC 30 as applicable design criteria for the IPS to support NSRST PSFs 1.1.3, 1.3.2, and 2.2.1. In addition, LME identified PDC 1 – 6 as cross-cutting and/or programmatic design requirements with which the IPS design complies. The NRC staff finds that these PDC selections are consistent with PSAR section 5.3 and are sufficient to ensure that the design of the IPS meets the RSFs.

Based on the PDC and safety function evaluations, described above, the NRC staff finds that the preliminary design of the IPS provides adequate support for NSRST PSFs 1.1.3, 1.3.2, and 2.2.1.

Special Treatments

PSAR section 7.3.11.3 states that final special treatments for the IPS will be identified and selected following expansion of the design and PRA to align with the development of system-level reliability and capability targets, consistent with the process described in NEI 18-04, as endorsed by RG 1.233. A general description of special treatments applied to NSRST SSCs is described in PSAR section 7.2 and evaluated by section 7.2 of the SE.

The programs described in PSAR chapter 8 and chapter 12 are preliminarily identified in the PSAR as programmatic special treatments to be applied to NSRST SSCs. The NRC staff finds that while specific special treatments for IPS components have not yet been fully identified, the preliminary special treatments are provided through compliance with PDC 6. The special treatments include preoperational testing and startup testing, described in PSAR sections 12.1.1.2 and 12.1.1 respectively. It also includes the RIM and IST programs and the NRC staff will evaluate the operational program scoping of SSCs during the OL application based on final risk-significance determinations. The LME QAPD also addresses QA for NSRST SSCs.

PSAR section 7.3.11.3 states that no IPS components have been identified as incorporating FOAK technology, design, or configurations that would warrant FOAK special treatment testing.

Based on the NRC staff's review of the preliminary special treatments described in the PSAR, the NRC staff finds that these special treatments are appropriate for the PSAR stage. A final determination of sufficiency of special treatments applied to the IPS system will be performed at OL application stage.

Technical Specifications

The applicant identified certain IPS functions for TS. The proposed list of preliminary LCOs identified in table IV-3 of the TS for the IPS lists:

- Field transmitters and process sensors
- Components that provide signal conditioning and setpoint comparison
- Components that perform logic functions

The NRC staff reviewed information provided in the PSAR related to the proposed parameters. The proposed LCOs will ensure that the IPS can perform the PSFs and provide DID for the RPS. The NRC staff finds that the information provided is compliant with 10 CFR 50.34(a)(5) given the variables, conditions, or items that are likely to become subjects of TS for the facility are identified.

7.3.11.2 Conclusion

Based on the review described above, the NRC staff concludes that the information in PSAR section 7.3.11 is sufficient and meets the applicable guidance and regulatory requirements identified in this SE section and at the beginning of this chapter for the issuance of a CP. Specifically, the information provided:

- meets the requirements of 10 CFR 50.34(a)(3)(i) for the IPS by describing the preliminary design identifying the applicable PDC,

- meets the requirements of 10 CFR 50.34(a)(3)(ii) by describing how the IPS design is consistent with the applicable PDC,

- meets the requirements of 10 CFR 50.34(a)(3)(iii) by providing the materials of construction and general arrangement of the IPS, where available, and

- meets the requirements of 10 CFR 50.34(a)(5) by identifying preliminary IPS variables and conditions that are subjects of TS.

7.3.12 Distributed Control Systems

PSAR section 7.3.12 describes the DCS. The DCS is a subsystem of the PCDAS and is a digital I&C system that provides a means for receiving information from field sensors, executing both automatic and manual control logic, interfacing with field actuators, and operator interface. The DCS provides DID by supporting NSRST PSFs to minimize the frequency of challenges to the RPS.

PSAR section 7.3.12 states that the DCS comprises multiple levels of control architecture, including input-output (I/O) modules, process controllers, application servers, and human-machine interfaces (HMIs). The process controllers execute simple and advanced controls, permissives, interlocks, and the controls which transition supported systems between the specified operating states.

The DCS comprises the following subsystems, which collectively provide the means for plant monitoring and operation via HMIs:

- Operator display system
- Operator alarm system
- Operator procedure system

The CEB contains the centralized DCS modules in the main control room (MCR), which provides the operator interface for control and monitoring. As the DCS is an integrated plant system, the DCS equipment is located throughout the NI and conventional island. Operation of the DCS is independent from, and cannot interfere with, operation of the RPS. The DCS is interlocked with the RPS and IPS, such that the RPS and IPS protective actions both take priority over all DCS functions. The DCS control functionality is enabled only if the RPS and IPS are in the normal operational states. Electric power is needed for the DCS to support NSRST PSFs.

In addition to PDCs 1-6, the following PDCs are applicable to the DCS:

- PDC 10 - Reactor design
- PDC-CDC 13 - Instrumentation and control
- PDC 19 - Control room
- PDC-CDC 34 - Residual heat removal
- PDC-CDC 44 - Structural and equipment cooling

7.3.12.1 Technical Evaluation

Safety Significance

The DCS is a subsystem of the PCDAS and is classified as NSRST. The DCS provides DID by performing two NSRST PSFs to minimize the frequency of challenges to the RPS and is summarized by the LBEs identified in PSAR table 5.5-10. Specifically, the DCS can restore active heat removal by control of the HCS, SSS, and RBCW functions that restore and maintain active heat removal. The DCS can also control heat removal of the reactor cavity through active means by control of the active RCCS functions that control heat removal from the reactor cavity.

As confirmed by the NRC staff via RCI (ML26239A157), LME is in the process of incorporating the results of the updated risk-significance evaluations following the NEI 18-04 methodology. The preliminary results of these evaluations indicate that the DCS is risk significant, which would mean that failure of the prevention or mitigation function of the DCS could lead some of the LBEs crediting DCS to exceed the F-C target.

In section 5.5 of the SE, the NRC staff determined that the classification of the DCS as NSRST is consistent with the guidance in NEI 18-04, as endorsed with clarifications by RG 1.233. Based on the information provided in the PSAR, the NRC staff finds that the DCS classification as NSRST is reasonable.

Compliance with Applicable PDCs

Quality Standards and Records (PDC 1)

The NRC staff reviewed PSAR table 7.3.12-1, which identifies the preliminary consensus codes and standards applicable to the DCS. These codes and standards address areas including quality and equipment qualification. Specifically, IEC 61508, "Functional Safety of Electrical/Electronic/Programmable Electronic Safety-Related Systems," is listed as one of the applicable standards. While IEC 61508 is a non-nuclear specific standard and has not been reviewed or endorsed by the NRC staff for use in SR I&C, it is an internationally recognized, risk-informed, performance-based standard developed for industrial safety applications. Given the NSRST classification, use of IEC 61508 is reasonable for the DCS. Additionally, the DCS is designed, fabricated, constructed, and tested in accordance with the QAP described in part III of the LMGS QAPD and PSAR section 8.1. Additional special treatments related to QA 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 QA requirements under PDC 1 for the NSRST portions of the DCS. 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 NRC staff's review of the information provided in the PSAR regarding codes and standards applicable to the DCS and the evaluation in section 8.1, the NRC staff finds that the preliminary design of the DCS is consistent with PDC 1.

Design Basis for Protection Against Natural Phenomena (PDC 2)

The NRC staff reviewed PSAR section 7.3.12.1, which states that the DCS is designed to withstand the effects of natural phenomena and the determination of associated DBHLs described in PSAR sections 6.1.1 and 6.1.3 and evaluated in section 6.1 of this SE. DBHL related requirements may be applied to the NSRST DCS to prevent interactions with SR SSCs that would impact the ability of the SSC to perform an RSF. The DCS is designed to withstand seismic events and achieve performance levels corresponding with its designated seismic category, which is shown in table 3.7-2 and as described in PSAR section 3.7.

Also stated in PSAR section 7.3.12.1, the DCS components required to support NSRST PSFs are physically located in the MCR and co-located within the structures containing the controlled equipment throughout the NI. The special treatment process, described in PSAR section 4.2 and evaluated in section 4.2 of this SE, determines the specific hazard event frequencies and any resulting design requirements necessary for the DCS. Based on the NRC staff's review of the information provided in the PSAR on how the IPS is designed to withstand the effects of natural phenomena and the evaluations in sections 3.7, 4.2, and 6.1, the NRC staff finds that the preliminary design of the DCS is consistent with PDC 2.

Fire Protection (PDC 3)

The NRC staff reviewed PSAR section 7.3.12.1, which states that the DCS is designed, located, and configured to minimize the probability and effect of fires and explosions through the selection of noncombustible and non-fire sustaining materials, to the extent practical. Further, the DCS is protected by fire protection systems designed, installed, and tested in accordance with applicable NFPA standards and as determined by the FHA and FSSA to be necessary to protected safety-significant SSCs and to ensure performance of safe-shutdown functions. These systems are a component of the comprehensive plant FPP and are designed to meet the

requirements of 10 CFR 50.48(a)(1) and are evaluated in section 8.6 of this SE. Based on the NRC staff's review of the information provided in the PSAR on the fire protection features designed to meet the requirements of 10 CFR 50.48(a)(1) and the evaluation in section 8.6 of this SE, the NRC staff finds that the preliminary design of the DCS is consistent with PDC 3.

Environmental and Dynamic Effects Design Bases (PDC 4)

The NRC staff reviewed PSAR section 7.3.12.1, which states that the DCS is designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and applicable AOOs. The DCS components are separated by train as necessary and protected from dynamic effects that may result from equipment failures and from external hazards. As stated in PSAR section 6.1.3 and evaluated by the NRC staff in section 6.1 of this SE, DBHL related requirements are applied to the NSRST IPS to prevent interactions with SR SSCs that would impact the ability of the SSC to perform an RSF. Based on the NRC staff's review of the information provided in the PSAR on the DCS design consideration of environmental conditions and effects and the evaluation in section 6.1, the NRC staff finds that the preliminary design of the DCS is consistent with PDC 4.

Sharing of Structures, Systems, and Components (PDC 5)

The NRC staff reviewed PSAR section 7.3.12.1, which states the DCS is shared between reactor modules. The sharing of integrated controls equipment between reactor modules will not significantly impair the ability of the DCS to perform NSRST PSFs, including in the event of an AOO, DBE, or DBA in one reactor module, an orderly shutdown of the remaining units. The DCS is described to have redundant I/O modules, controllers, and logic between reactor modules and between redundant trains for the supported NSRST PSFs. No reactor modules or trains share common I/O or controllers. Based on the NRC staff's review of the information provided in the PSAR on shared and non-features of the DCS via redundant features, the NRC staff finds that the preliminary design of the DCS is consistent with PDC 5.

Monitoring, Inspection, Testing, Surveillance (PDC 6)

As discussed in the safety significance subsection above, LME is in the process of incorporating the results of updated risk-significance evaluations following the NEI 18-04 methodology. The preliminary results of these evaluations indicate that the DCS is risk significant. 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 during the OL application based on final risk-significance determinations. The NRC staff reviewed PSAR section 7.3.12.1, which states that the safety-significant DCS SSCs are designed with indications from online monitoring techniques for operators. The IPS also meets applicable portions of PDC 6 through a combination of preservice and inservice requirements. Safety-significant IPS SSCs undergo periodic surveillances. Additional preoperational and startup testing is described in PSAR sections 12.1.1.2 and 12.1.1 and evaluated in section 12.1 of this SE. Baseline testing along with monitoring, testing, and surveillance requirements provide confidence that the capability of the DCS is retained through plant life. Based on the NRC staff's review of the information provided in the PSAR on the RIM and IST programs evaluated in sections 8.2 and 8.4, online monitoring capabilities, application of preservice and inservice requirements, and the testing requirements evaluated in section 12.1, the NRC staff finds that the preliminary design of the DCS is consistent with PDC 6.

Reactor Design (PDC 10)

The NRC staff reviewed section 7.3.12.1, which states that PDC 10 is a higher-level requirement that is met in the DCS design via application of PDC-CDC 34. The NRC staff finds that this statement is appropriate because it is consistent with the NRC-approved Xe-100 PDC TR. As discussed in the basis for PDC 10 in the Xe-100 PDC TR, PDC 10 is a design limit that supports acceptance criteria in PDC-CDC 34. As discussed in the basis for PDC 10 in the Xe-100 PDC TR, PDC 10 is a design limit that support PDCs. Therefore, PDC 10 requirements are satisfied by meeting PDC-CDC 34. The evaluation for compliance of the DCS design to PDC-CDC 34 is performed later in this section.

Instrumentation and Control (PDC-CDC 13)

The NRC staff reviewed PSAR section 7.3.12.1, which states that the DCS 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 PSAR also states that the DCS monitors plant parameters by collecting information from field sensors and communications from other I&C components and initiates automatic control functions when monitored parameters reach or exceed specified setpoints, to maintain variables within prescribed operating ranges. Based on the NRC staff's review of the information provided in the PSAR, the NRC staff finds the preliminary design of the DCS to have I&C necessary to accomplish PSFs and to support overall DID is consistent with PDC-CDC 13.

Control Room (PDC 19)

The NRC staff reviewed PSAR section 7.3.12.1, which states that the DCS is designed to provide HMIs in the MCR from which actions can be taken to operate the plant safely during normal conditions and to maintain it in a safe condition during certain LBEs. HMIs are designed in accordance with the human factors engineering program, described in PSAR section 8.7 and evaluated in section 8.7 of this SE. Based on the NRC staff's review of the information provided in the PSAR on the HMIs providing the capability to operate the plant and the evaluation in section 8.7, the NRC staff finds that the preliminary design of the DCS is consistent with PDC 19.

Residual Heat Removal (PDC-CDC 34)

The NRC staff reviewed PSAR section 7.3.12.1, which states that the DCS 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 an ultimate heat sink at a rate such that SARRDLs are not exceeded during AOOs. The PSAR also states that the DCS is designed to perform this function by monitoring plant parameters and providing control functions for the HCS, SSS, RCCS, RBCW, NIE and NIHV by initiating automatic control functions when associated parameters reach or exceed specified setpoints and also allowing manual initiation of control functions. Based on the NRC staff's review of the information provided in the PSAR on the DCS support of active residual heat removal through control of safety-significant SSCs to accomplish PSFs, the NRC staff finds that the preliminary design of the DCS is consistent with PDC-CDC 34.

Structural and Equipment Cooling (PDC-CDC 44)

The NRC staff reviewed PSAR section 7.3.12.1, which states that the DCS is designed to support active transfer of heat from safety-significant SSCs to a heat sink, as necessary, to transfer the combined heat load during certain LBEs. The PSAR also states that the DCS is designed to perform this function by monitoring plant parameters and providing control functions for the RBCW, NIE, and NIHV by initiating automatic control functions when associated parameters reach or exceed specified setpoints and also allowing manual initiation of control functions. Based on the NRC staff's review of the information provided in the PSAR on the DCS support of active transfer of heat for structural and equipment cooling to accomplish PSFs, the NRC staff finds that the preliminary design of the DCS is consistent with PDC-CDC 44.

Safety Functions

PSAR section 7.3.12.1 states that the DCS is designed to support the NSRST PSF 1.2.2.2 to restore active heat removal and PSF 1.2.2.3 to control heat removal of the reactor cavity through active means.

LME identified PDC-CDC 13, PDC-CDC 34, and PDC-CDC 44 as applicable design criteria for the DCS to support NSRST PSFs 1.2.2.2 and 1.2.2.3. In addition, LME identified PDCs 1 - 6 as cross-cutting and/or programmatic design requirements with which the DCS design complies. The NRC staff finds that these PDC selections are consistent with PSAR section 5.3 and are sufficient to ensure that the design of the DCS meets the RSFs.

Based on the PDC and safety function evaluations, described above, the NRC staff finds that the preliminary design of the DCS provides adequate support for NSRST PSFs 1.2.2.2 and 1.2.2.3.

Special Treatments

PSAR section 7.3.12.3 states that final special treatmentsfor the DCS will be identified and selected following expansion of the design and PRA to align with the development of system-level reliability and capability targets, consistent with the process described in NEI 18-04, as endorsed by RG 1.233. A general description of special treatments applied to NSRST SSCs are described in PSAR section 7.2 and evaluated by section 7.2 of the SE.

The programs described in PSAR chapter 8 and chapter 12 are preliminarily identified in the PSAR as programmatic special treatments to be applied to NSRST SSCs. The NRC staff finds that while specific special treatments for DCS components have not yet been fully identified, the preliminary special treatments are provided through compliance with PDC 6. The special treatments include preoperational testing and startup testing, described in PSAR sections 12.1.1.2 and 12.1.1 respectively. It also includes the RIM and IST programs, evaluated in sections 8.2 and 8.4 and the NRC staff will evaluate the operational program scoping of SSCs during the OL application based on final risk-significance determinations. The LME QAPD also addresses QA for NSRST SSCs.

PSAR section 7.3.12.3 states that no DCS components have been identified as incorporating FOAK technology, design, or configurations that would warrant FOAK special treatment testing.

Based on the NRC staff's review of the preliminary special treatments described in the PSAR, the NRC staff finds that these special treatments are appropriate for the PSAR stage. A final determination of sufficiency of special treatments applied to the DCS system will be performed at OL application stage.

7.3.12.2 Conclusion

Based on the review described above, the NRC staff concludes that the information in PSAR section 7.3.12 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 the requirements of 10 CFR 50.34(a)(3)(i) for the DCS by describing the preliminary design identifying the applicable PDC,

- meets the requirements of 10 CFR 50.34(a)(3)(ii) by describing how the DCS design is consistent with the applicable PDC, and

- meets the requirements of 10 CFR 50.34(a)(3)(iii) by providing the materials of construction and general arrangement of the DCS, where available.

8 PLANT PROGRAMS

This chapter of the safety evaluation (SE) describes the U.S. Nuclear Regulatory Commission (NRC) staff's review and evaluation of the Long Mott Generating Station's (LMGS) preliminary safety analysis report (PSAR) chapter 8, "Plant Programs," which provides an overview of plant programs relied upon to support the Nuclear Energy Institute (NEI) 18-04 process, addressing the purpose, scope, and performance objectives as well as applicability to safety-related (SR) structures, systems, and components (SSCs), non-safety-related with special treatment (NSRST) SSCs, and operations activities. The PSAR states that details of operational programs and special treatments will be provided in the final safety analysis report (FSAR) as part of the application for an operating license (OL). Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(b)(6) requires information concerning facility operations to be included in the FSAR that a holder of a construction permit (CP) would submit when requesting an OL. Accordingly, the NRC staff's review of operational programs to be used to assure safe operation would take place as part of the NRC staff's review of the OL application.

8.1 Quality Assurance Program and Reliability Assurance Program

PSAR section 8.1 describes the quality assurance program (QAP) and references topical report (TR), "Long Mott Energy, LLC Quality Assurance Program Description," (hereafter referred to as the Long Mott Energy (LME) quality assurance program description (QAPD)) (ML24345A043).

PSAR section 8.1.4 describes the LMGS reliability assurance program (RAP) that applies to SR and NSRST SSCs.

The applicable regulatory requirements for the evaluation of the QAP and RAP are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a)(7); and
- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,"

The applicable guidance for the evaluation of QAP and RAP is as follows:

- Division of Advanced Reactors and Non-Power Production and Utilization Facilities (DANU) Interim Staff Guidance (ISG) DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications - Roadmap," section 4, "Quality Assurance Plan," (ML23277A139).
- Regulatory Guide (RG) 1.233, Revision 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 NEI 18-04, Revision 1, "Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," (ML19241A472).
- NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," section 17.4, Revision 1, "Reliability Assurance Program,"

(ML13296A435), and section 17.5, Revision 1, "Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants," (ML15037A441).

- Staff Requirements Memorandum (SRM) SRM-SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs."

PSAR section 5.3 provides definitions of principal design criteria (PDCs). The NRC staff identified the QAP as a key part of meeting PDC 1, "Quality Standards and Records." PDC 1 is applicable to all safety-significant SSCs (i.e., SR and NSRST SSCs) and how LME has applied the QAP in meeting PDC 1 is described in PSAR sections 6.4 and 7.3 and evaluated in sections 6.4 and 7.3 of this SE.

8.1.1 Technical Evaluation

8.1.1.1 Quality Assurance Program

The NRC staff reviewed PSAR section 8.1 and the LME QAPD, which is incorporated by reference in PSAR table 1.4-4. The NRC staff approved the LME QAPD for use with a CP application subject to two limitations (ML25269A117). LME addressed the limitations associated with QAPD TR in PSAR table 1.4-8. The NRC staff's evaluation of PSAR table 1.4-8 is presented in SE section 1.4.

The LME QAPD established the QAP at LMGS. Within the QAP, the SSC are divided into categories based on safety classification (e.g., SR, NSRST) which determines the treatment of the safety-significant SSC. The NRC staff will verify implementation of the QAP as part of the NRC advanced reactor construction oversight program.

PSAR section 8.1.2 summarizes SSC classification and associated quality assurance (QA) guidance provided in NEI 18-04. The NRC staff finds this summary acceptable because it is consistent with NEI 18-04, which is endorsed by RG 1.233.

PSAR section 8.1.3 summarizes QAP requirements and states, in part, that "SR SSCs QA requirements are consistent with the LME QAPD, Part I and Part II" and "NSRST SSCs QA requirements are consistent with the LME QAPD, Part III. Additional QA requirements, up to and including NQA-1, may be applied to NSRST SSCs as a special treatment consistent with non-safety-related QA program guidance through the NEI 18-04 Integrated Decision-Making Process (IDP)." Since the QA treatment for SR SSCs is consistent with 10 CFR Part 50, Appendix B, and QA treatment for NSRST SSCs is at a minimum consistent with the LME QAPD and expanded with NEI 18-04 special treatments, as endorsed by RG 1.233, the NRC staff determined that the QAP requirements summarized in PSAR section 8.1.3 are acceptable.

8.1.1.2 Reliability Assurance Program

The NRC staff reviewed PSAR section 8.1.4, which describes the RAP, in accordance with the associated acceptance criteria within the context of applicability to LMGS' specific reactor technology in SRP 17.4. Specifically, the NRC staff reviewed the RAP to determine whether it adequately considers the following attributes:

- Describe the RAP used during the design including its scope, purpose, and objectives.

- Evaluate and prioritize the SSCs in the design, based on their degree of risk significance.
- List SSCs designated as risk significant.
- For those SSCs designated as risk significant, determine:
    - Dominant failure modes that consider industry experience and analytical models.
    - Key assumptions and risk insights from probabilistic, deterministic, or other methods that consider operations, maintenance, and monitoring activities.

The NRC staff determined that the applicant included a description of the details of the design reliability assurance program (D-RAP) that will be implemented during the design, procurement, fabrication, construction, and preoperational testing activities preceding initial fuel load. The applicant's description included a discussion of the scope, purpose, objectives, framework, and those activities that will occur in the two phases of the D-RAP (i.e., detailed plant design and detailed plant design following issuance of an OL). The NRC staff confirmed that the scope, purpose, and objectives of the applicant's D-RAP provide sufficient detail for the preliminary design and are consistent with subsection I of SRP section 17.4, Revision 1. Based on its review, the NRC staff finds the description of the D-RAP to be acceptable.

The NRC staff determined that the applicant has established and applied the appropriate D-RAP programmatic controls to support design and construction activities. Consistent with guidance in SRP section 17.4, Revision 1, the programmatic controls established by the applicant address organization responsibilities, design control activities, procedures and instructions, records control, corrective actions, and audit plans. Based on its review, the NRC staff finds that the applicant has established adequate programmatic controls for the D-RAP.

The NRC staff determined that the applicant's methodology for identifying the SSCs within the scope of the RAP is based on risk-informed performance-based methodology for design and licensing basis development from NEI 18-04. The methodology provides an objective and repeatable approach to identify SSCs within the scope of the RAP. The applicant's methodology is consistent with the practice for identifying risk-significant SSCs, as discussed in RG 1.233. This methodology is consistent with that described in SRP section 17.4, Revision 1. Based on the NRC staff's review, the NRC staff finds that the applicant has established an adequate methodology for identifying RAP SSCs.

The NRC staff determined that the applicant uses an integrated decision-making process (IDP) and the integrated decision-making process panel (IDPP) as described in NEI 18-04, to evaluate the adequacy of defense-in-depth (DID) and the safety classification of SSCs, which includes determining which SSCs are categorized as NSRST based on risk significance. The roles and responsibilities, decision-making attributes, and actions to establish DID from the IDPP and the IDP are described in NEI 18-04, Section 5.9, "Risk-Informed and Performance-Based Evaluation of DID Adequacy." The NRC staff verified that the description of the IDPP is consistent with that described in SRP section 17.4, Revision 1. Based on the NRC staff's review, the NRC staff finds the description and implementation of the IDPP to be acceptable.

Overall, the NRC staff determined that the application contains a comprehensive list of SSCs subject to RAP requirements consistent with the state of the plant design and probabilistic risk assessment (PRA) at the time of CP application and the list of RAP SSCs based on the methodology for identifying and characterizing SSCs is within the scope of the RAP. The lists in PSAR chapter 5, "PRA Safety Functions, Design Criteria, and SSC Classification," sufficiently identify the RAP SSCs classified as SR and NSRST that are needed to meet risk targets, and PSAR chapter 6, "Safety-related SSC Criteria and Capabilities," and chapter 7 "Non-Safety-Related with Special Treatment Structures, Systems, and Components Criteria and Capabilities," include the basis for including each SSC in the RAP, for SR and NSRST SSCs. Consistent with SRP section 17.4, Revision 1, the NRC staff finds that the applicant appropriately cited PSAR chapters 6 and 7 that describe the SSC boundaries. Based on the NRC staff's review, the NRC staff finds the RAP SSCs list to be acceptable.

The NRC staff determined that the application describes a proposed process for determining dominant failure modes of RAP SSCs. Consistent with guidance in SRP section 17.4, Revision 1, the applicant's proposed process for determining dominant failure modes incorporates consideration of insights from the PRA model, analytical methods, and industry experience. The process described by the applicant is consistent with the guidance for an acceptable process described in SRP section 17.4, Revision 1, and therefore is acceptable.

The NRC staff determined that the applicant specified where the QA controls for design activities are described. The applicant cited the LME QAPD, which describes QA controls applied to non-safety-related RAP SSCs. The QA controls are evaluated in the NRC staff's SE of LME QAPD. These controls are consistent with the NRC staff's expectations for QA controls described in SRP section 17.4, Revision 1, and are therefore acceptable.

8.1.2 Conclusion

Based on the NRC staff's review, the NRC staff finds that the QAP, as described in PSAR section 8.1 and in the LME QAPD, is consistent with the applicable regulatory guidance and, therefore, meets the regulatory requirements identified in 10 CFR 50.34(a)(7) for the issuance of a CP. The NRC staff reviewed PSAR section 8.1.4 in accordance with Item E of SECY-95-132 and SRP section 17.4, Revision 1, and concludes that the D-RAP program as described in the PSAR is acceptable to demonstrate that it follows SRP section 17.4, Revision 1 because the plant is designed, to be constructed, and to be operated in a manner that is consistent with the risk insights and key assumptions from the probabilistic and deterministic analyses.

8.2 Reliability and Integrity Management Program

PSAR section 8.2 describes the reliability and integrity management (RIM) program which will be implemented in accordance with American Society of Mechanical Engineers (ASME) Section XI, Division 2, "Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants," as endorsed in RG 1.246, "Acceptability of ASME Code, Section XI, Division 2, 'Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants,' for Non-Light Water Reactors," Revision 0 (ML22111A087). The objectives of the RIM program are to define, evaluate, and implement strategies to ensure that the facility risk and reliability goals (i.e. reliability targets) for each passive SSC whose failure could adversely affect facility safety and reliability are defined, achieved, and maintained throughout the facility

lifetime through a combination of monitoring, examination, tests, operation and maintenance programs.

The applicable guidance for the evaluation of RIM is as follows:

- RG 1.246

The NRC staff identified the RIM program as a key part of meeting the following PDCs. How LME has applied the RIM program in meeting these PDCs for specific SSCs is described in PSAR sections 6.4 and 7.3 and evaluated in sections 6.4 and 7.3 of this SE. Since there are no requirements for inservice inspection (ISI) applicable to CPs, these PDCs are not addressed below:

- PDC 4, “Environmental and Dynamic Effects Design Basis”;
- PDC 6, “Monitoring, Inspection, Testing, and Surveillance”; and
- PDC 14, “Reactor Helium Pressure Boundary.”

8.2.1 Technical Evaluation

The NRC staff reviewed PSAR section 8.2, in which the applicant stated that the RIM program will be developed following the 2019 Edition of ASME Boiler and Pressure Vessel Code Section XI, Division 2 (ASME.XI.2) and will be implemented in accordance with RG 1.246, which endorses the use of RIM. In addition, the RIM program will follow the associated code case N-954-1, “Requirements for Monitoring and Nondestructive Examination (MANDE) of Graphite Core Components in a High-Temperature Gas Reactor.” Code Case N-954-1 is not endorsed in the current revision of RG 1.246. The NRC staff reviewed the approaches in N-954-1 to develop the MANDE for graphite core components. LME's approach is acceptable because N-954-1 provides technically sound guidance on establishing appropriate MANDE for degradation mechanisms relevant to graphite materials in high-temperature gas reactors (HTGRs).

As described in ASME.XI.2, the RIM program scope shall include applicable SSCs whose failure could adversely affect plant safety and reliability. The RIM program requires conducting a degradation mechanism assessment (DMA) for each component in the scope of the RIM program, determining the reliability targets for the SSCs, and identifying RIM strategies to ensure reliability targets can be met. The objective of the RIM program is to ensure that passive components within the RIM program scope achieve an acceptable level of reliability to support the plant PRA throughout the life of the plant. The RIM program involves design interaction, performance monitoring, inspections, tests, maintenance, and replacements, as strategies to ensure the SSCs achieve an acceptable level of reliability to support the LMGS PRA.

The NRC staff reviewed the applicant’s preliminary RIM program scope and the criteria used by the applicant to determine which SSCs are included in the RIM program as described in PSAR section 8.2. The applicant states that the RIM program scope includes SR and risk-significant NSRST passive SSCs but does not include NSRST SSCs relied on for DID The preliminary RIM program scope departs from the scope requirement in ASME BPVC XI-2 that requires passive SSCs whose failure could adversely affect plant safety and reliability to be included in the RIM program; under ASME BPVC XI-2, NSRST SSCs relied on for DID would be included. The NRC

staff did not make any findings on the program scope but notes that a final program scope, including a basis for the departure from ASME BPVC XI-2, as endorsed by RG 1.246, will be reviewed at the OL application stage.

The NRC staff reviewed PSAR section 8.2.2 which describes the preliminary DMA. The applicant states that considerations such as material selection, service conditions, and operating environment will be evaluated. Existing operational and test data is used to inform material selection. ASME XI-2 Article VII-3, "High Temperature Gas-Reactor-Type Facilities," provides requirements for identifying and evaluating potentially active degradation mechanisms specific to HTGR facilities. In addition, while not explicitly stated in the PSAR section 8.2.2, the results of the materials qualification program described in PSAR section 3.8, and the research and development program described in chapter 13, "Research and Development Program," will also inform the DMA and associated selection of MANDE strategies. The NRC staff did not make any findings on the preliminary considerations for the DMA because 10 CFR 50.34(b)(6) requires information concerning facility operations to be included in the FSAR that a holder of a CP would submit when requesting an OL. The NRC staff notes that ASME XI-2.VII-3 is approved for evaluating the DMA, with conditions, in RG 1.246.

The NRC staff did not make findings on the remainder of PSAR section 8.2, including: (1) the acceptability of the reliability target allocation or confirmation process of the preliminary reliability targets, (2) preliminary RIM strategies and evaluation of uncertainties, or (3) preservice inspection. The NRC staff expects the applicant to provide the final reliability targets and RIM strategies for SSCs that are in the RIM program, including the technical basis on how the reliability targets are derived and met, and consideration of uncertainties, in the FSAR supporting an OL application.

8.2.2 Conclusion

The NRC staff will review the RIM program at the OL application stage as required by 10 CFR 50.34(b)(6)(iv). Due to the preliminary nature of the program, the applicant is not seeking to demonstrate compliance with this regulation at the CP application stage. The NRC staff is not making any regulatory findings pertaining to the contents of the RIM program meeting the requirements of ASME BPVC XI-2, as conditioned in RG 1.246 because 10 CFR 50.34(b)(6) requires information concerning facility operations to be included in the FSAR that a holder of a CP would submit when requesting an OL. The NRC staff will evaluate the RIM program 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, which includes NSRST SSCs relied on for DID.

8.3 Maintenance Program

PSAR section 8.3 provides a high-level description of the maintenance program. The stated objectives of the program are to control the performance of maintenance activities in the plant and the program is applied to passive and active safety-significant SSCs.

PSAR section 8.3 describes the program for maintenance activities for passive and active safety-significant SSCs addressed by 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The NRC staff notes that the applicant has identified appropriate guidance documents for maintenance activities. However, the NRC

staff is not making any regulatory findings in this SE because, per 10 CFR 50.34(b)(6)(iv), plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of SSCs are applicable at the OL application stage. For this reason and because the applicant is not seeking to demonstrate compliance with this regulation at the CP application stage consistent with 10 CFR 50.34(a)(7), the NRC staff is not making any regulatory findings pertaining to this section.

While the NRC staff did not make any regulatory findings pertaining to the maintenance program described in the PSAR, the NRC staff reviewed the approach discussed in developing the maintenance program and has the following observations. The NRC staff notes that LME is using the maintenance program in lieu of the inservice testing (IST) program for NSRST SSCs relied on for DID but has not provided information on how diagnostic testing would be performed. In addition, while an SSC could be in scope for both the IST and maintenance programs under 10 CFR 50.65, the regulations and purpose of the programs are separate. The NRC staff will evaluate the maintenance program as part of the OL application review, including the basis and acceptability of having separate IST programs as well as the program for performing diagnostic testing of NSRST SSCs relied on for DID.

8.4 Inservice Testing Program

The NRC staff reviewed PSAR section 8.4, which states that an IST program will be established to meet the intent of the requirements in 10 CFR 50.55a(f). The PSAR states that the IST program shall “fulfill the intent of the requirements of 10 CFR 50.55a(f)” and the “IST program is implemented in accordance with ASME Operation and Maintenance OM-2 Code, “Component Testing Requirements at Nuclear Facilities,”” 2024 Edition (OM-2-2024). The PSAR states that the IST program is applied to active SR and risk-significant NSRST SSCs. In RG 1.220, “Acceptability of ASME OM-2 Code, Component Testing Requirements at Nuclear Facilities” (ML25329A088), the NRC staff endorsed ASME OM-2-2024 as an acceptable method for implementing an IST program. While OM-2 does not specify a scope for the IST program, the NRC staff notes that it may be applied to any active SSCs and allows for a graded IST approach. The NRC staff notes that the applicant has chosen to use the maintenance program rather than the IST program to ensure functional capability of safety-significant SSCs for active components. The NRC staff also noted that the maintenance program activities may include exercise and diagnostic testing which traditionally belong to the IST program. The NRC staff is not making any regulatory findings in this SE pertaining to this section or the acceptability of the applicant’s approach to use its maintenance program to meet the requirements for an IST program, as required by 10 CFR 50.55a(f) for an OL, because the requirements in 10 CFR 50.34(b)(6)(iv) are not applicable to a CP application and the applicant is not seeking to demonstrate compliance with this regulation at the CP application stage.

8.5 Environmental Qualification Program

The NRC staff reviewed PSAR section 8.5, which describes the program for qualification of the safety-significant electrical equipment in systems addressed by 10 CFR 50.49(b). The NRC staff notes that the applicant has identified appropriate guidance documents for environmental qualification of safety-significant electrical equipment. However, the NRC staff is not making any regulatory findings pertaining to this section because, per 10 CFR 50.49(a), the requirements in 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,” are not applicable to a CP application and the applicant is not seeking to

demonstrate compliance with this regulation at the CP application stage (i.e., the application does not include all the information required by 10 CFR 50.49). PDC 4, "Environmental and Dynamic Effects Design Basis," states that: "safety-significant structures, systems, and components shall be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences, design basis events, and design basis accidents commensurate with the safety significance of the functions to be performed." The NRC staff's evaluations of the design bases for SR and NSRST SSCs, with respect to addressing the requirements of PDC 4, are provided in chapters 6 and 7 of this SE.

8.6 Fire Protection Program

The NRC staff reviewed the preliminary design of the fire protection program (FPP) described in the LMGS PSAR to assess the adequacy of the design in support of the issuance of a CP.

The applicable regulatory requirements for the evaluation of fire protection for a CP are:

- 10 CFR 50.34(a)(3)(i).

The applicable guidance for the evaluation of fire protection is:

- RG 1.189, "Fire Protection for Nuclear Power Plants," Revision 5 (ML23214A287); and
- RG 1.191, "Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown." Revision 1 (ML20287A199).

Definitions of PDCs are included in PSAR section 5.3. The NRC staff identified the FPP as a key part of meeting PDC 3, "Fire Protection." How LME has applied the FPP in meeting PDC 3 for specific SSCs is described in PSAR sections 6.4 and 7.3 and evaluated in sections 6.4 and 7.3 of this SE.

8.6.1 Technical Evaluation

The NRC staff evaluated the sufficiency of the facility's preliminary fire protection systems and programs as described in the PSAR as required by PDC 3.

The NRC staff reviewed PSAR section 1.1.4.3.9, which states that the fire protection systems consist of:

- the nuclear island fire protection (NIFP) system, which provides fire protection for SSCs located within nuclear island buildings and structures;
- the conventional island fire protection (CIFP) system, which provides fire protection for the conventional island SSCs; and
- the plant fire protection system, which stores, supplies, and distributes fire water to the NIFP and CIFP systems.

The applicant states that these systems are one part of the overall FPP designed and maintained to meet the requirements of PDC 3 and that other parts of the FPP include

procedures, analyses, administrative controls, personnel, and other support programs. The NRC staff also reviewed PSAR section 1.1.4.4.10, in which the applicant describes the fire protection pump house and fire water storage tank structure. The applicant states that these structures provide adequate space for water supply and redundant water pumps for fire protection purposes at LMGS. The applicant also stated that the pump house contains the diesel fuel storage tank. The NRC staff evaluated the information in PSAR sections 1.1.4.3.9 and 1.1.4.4.10 and finds that the level of detail in these sections is acceptable for a preliminary design.

The NRC staff reviewed PSAR table 1.4-2 in which the applicant indicates that the final FPP will fully conform with the portions of RG 1.189 that are technically relevant to the applicant's non-light water reactor (non-LWR) design. Portions of the RG that are determined to be not technically relevant will have alternatives identified and justified in the FSAR. In PSAR section 8.6, the applicant states that deviations from RG 1.189 will be justified in the OL application. PSAR table 1.4-2 also indicates that the guidance provided in RG 1.191 will be addressed at the OL application stage. The NRC staff evaluated this information and determined that conformance with these RGs is not required during the preliminary design phase. Based on the NRC staff's evaluation, the NRC staff finds that the descriptions in the PSAR are acceptable because conformance with these RGs will be addressed at the OL application stage.

The NRC staff reviewed PSAR table 1.4-3 in which the applicant lists a selection of codes and standards related to fire protection and building construction that will be used for LMGS, including those from the American Society of Testing Materials, the Institute of Electrical and Electronics Engineers (IEEE), the National Fire Protection Association (NFPA), and Underwriters Laboratories. The NRC staff reviewed the table and determined that the applicant included well established consensus codes and standards related to fire protection and building construction that they intend to follow during design and construction of the facility. Based on the NRC staff's review, the NRC staff finds the information provided in the PSAR regarding codes and standards for a preliminary design to be acceptable.

The NRC staff reviewed PSAR section 6.1.1, in which the applicant states that the design basis internal fire hazard has not been evaluated for the preliminary design but will be evaluated at the OL application stage. The NRC staff evaluated this section of the PSAR and finds that this information is acceptable for a preliminary design because an evaluation of the design basis internal fire hazard is not required at the CP stage and the applicant states that the design basis internal fire hazards will be evaluated at the OL application stage.

The NRC staff reviewed the introductory sections of PSAR chapters 6 and 7, in which the applicant states that, consistent with PDC 3, safety-significant SSCs would be designed and located to minimize the probability and effect of fires and explosions consistent with other safety requirements and the safety significance of the functions to be performed. The applicant also states that non-combustible and fire-resistant materials will be used wherever practical throughout, particularly in locations with safety-significant SSCs. Finally, the applicant states that fire detection and fighting systems will be: of appropriate capacity and capability; provided and designed to minimize the adverse effects of fires on safety-significant SSCs commensurate with the safety significance of their functions; and designed to ensure that its rupture or inadvertent operation does not significantly impair the safety capability of these SSCs. The NRC staff evaluated this information and determined that the design and location of these SSCs

would meet the requirements of PDC 3. Based on its evaluation, the NRC staff finds that the information provided in the PSAR regarding the design and location of SSCs is appropriate for a preliminary design because the SSCs would conform to the requirements of PDC 3.

The NRC staff reviewed PSAR section 8.6 in which the applicant describes the elements of the FPP and states that the details will be developed for the OL application. These elements include:

- Fire detection and suppression systems;
- Plant and system design and layout, including use of fire-resistant assemblies (e.g., fire barriers, fire doors, fire dampers);
- Engineering analyses, including a fire hazards analysis and safe shutdown analysis;
- Administrative controls, policies and procedures;
- Organization and staffing; and
- Support programs (QA, inspection and testing, maintenance, and training).

The applicant further states that the operational FPP will be in conformance with the technically relevant portions of RG 1.189, Revision 5. The applicant also states that the FPP will conform with the provisions of NFPA Standard 804 that are technically relevant to post-fire safe shutdown capability and mitigation of a radiological release resulting from a fire. Finally, the applicant states that justification for any portions of the FPP that require deviation or alternate methodologies from RG 1.189 or NFPA 804, will be provided in the OL application.

The NRC staff evaluated the information provided by the applicant in PSAR section 8.6 and finds it acceptable for a preliminary design because the applicant provided a description of the FPP and how the FPP will comply with the available guidance. Additionally, PDC 3 ensures that the final design of the FPP will conform with the provisions of RG 1.189 that are technically relevant to post-fire safe shutdown capability and mitigation of a radiological release resulting from a fire at LMGS, and LME states that all deviations or alternate methodologies from RG 1.189 will be described and justified in the OL application.

8.6.2 Conclusion

Based on the NRC staff's evaluation of the information provided by the applicant in the PSAR, the NRC staff has determined that the level of detail provided regarding the FPP and fire protection systems, demonstrates an adequate basis for a preliminary design that meets PDC 3 and thus 10 CFR 50.34(a)(3)(i). Therefore, the NRC staff finds that the preliminary description of the FPP, as described in the PSAR, provides reasonable assurance that the applicable regulatory requirements and guidance for the issuance of a CP are met. The NRC staff will confirm that the final design conforms to this design basis during the evaluation of the applicant's FSAR.

8.7 Human Factors Engineering Program

PSAR section 8.7 describes the human factors engineering (HFE) program and associated technical information for the facility. PSAR section 8.7 describes an HFE program that would result in the following:

- Timeliness and high-quality performance of personnel tasks
- Support of personnel situation awareness through the development of human-system interfaces (HSI), procedures, staffing and qualifications, training, and management and organizational arrangements
- Acceptable balance of workloads for plant personnel and design of SSCs that support personnel in maintaining the reactor
- Development of HSIs that minimize human error and support error detection and recovery capability

The applicable regulatory requirements for the evaluation of the HFE are as follows:

- 10 CFR 50.34(a)(3)(i); and
- 10 CFR 50.34(f)[1]
    - (2)(iii); (ii); (iv); (v); (xvii); (xviii); (xix); and (xxvii)
    - (3)(i)

As discussed in SE section 1.4.1, the NRC staff's review includes evaluating compliance of the LMGS design with the technically relevant portions of 10 CFR 50.34(f), which are identified in PSAR table 1.4-1.

The applicable guidance for the evaluation of HFE are as follows:

- DANU-ISG-2022-05, "Advanced Reactor Content of Application Project Chapter 11, 'Organization and Human-System Considerations' Interim Staff Guidance," March 2024 (ML23277A143);
- NUREG-0711, "Human Factors Engineering Program Review Model," Revision 3, November 2012 (ML12324A013);
- RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," May 1973 (ML003740127); and

[1] As stated in DANU-ISG-2022-05, "The requirements here do not apply to 10 CFR Part 50 applicants not listed in 10 CFR 50.34(f). Rulemaking is underway to align the requirements of 10 CFR Part 50 and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," that may make these requirements applicable to all 10 CFR Part 50 applicants. In the interim, the NRC staff should propose license conditions on 10 CFR Part 50 applications to address these regulations."

- RG 1.232, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors,” Revision 0, April 2018 (ML17325A611).

The NRC staff identified the HFE program as a key part of meeting PDC 19, “Control Room.”

8.7.1 Technical Evaluation

The NRC staff evaluated PSAR section 8.7 using the guidance and criteria as described in section 11.1.3 of DANU-ISG-2022-05. The objective of this review was to ensure that the application includes sufficient information, of scope and level of depth appropriate for a PSAR, and to understand the development of the HFE program for the LMGS facility.

8.7.1.1 Inclusion and Application of Appropriate PDC

DANU-ISG-2022-05, Criterion 11.2(b) states, in part, that an application should describe how the HFE and planned human-system interactions satisfy the design-specific PDC related to HFE and human-system interactions. Paragraph 50.34(a)(3)(i) of 10 CFR requires an applicant for a CP to provide the PDC for the facility. PDC 19 applies to the main control room (MCR) design, which is applicable to the HFE and planned human-system interactions. The applicant states in PSAR section 8.7.5 that the HFE program is applicable to the design of the MCR. In PSAR section 6.4.13, the applicant also described the reactor protections system cabinets as an alternate location to safely shut down the reactor outside the MCR. The RPS cabinets contain manual initiation switches that provide operators with the ability to safely shut down the reactor. In PSAR section 1.1.4.3.4, the applicant also described that the plant monitoring computer will be available to operators to monitor plant parameters outside of the MCR. The NRC staff reviewed the MCR and alternate location related to PDC 19 in the PSAR and concludes that the applicant addresses how the planned HFE and HSI for the proposed MCR and alternate location satisfy the design-specific PDC consistent with the criterion presented in section 11.2(b) of DANU-ISG-2022-05. The NRC staff finds that the PSAR complies with 10 CFR 50.34(a)(3)(i), by including a PDC specific to the MCR design and alternate location used for safe shutdown of the reactor.

8.7.1.2 Technologically Relevant HFE-Related Requirements of 50.34(f)

Section 11.1.3 of DANU-ISG-2022-05 describes that all 10 CFR Part 50 applications should describe how the non-LWR facility will satisfy any requirements of 10 CFR 50.34(f), with areas that are relevant to the non-LWR technology design. PSAR section 8.7.2 describes the regulatory compliance of the facility being designed and developed in accordance with 10 CFR.34(f) requirements.

The NRC staff reviewed the PSAR section 8.7.2 in accordance with 10 CFR 50.34(f)(2)(iii) and finds that the MCR is being designed using state-of-the-art human factors principles as required by 10 CFR 50.34(f)(2)(iii). Additionally, the preliminary design and layout of the MCR is described in section 1.1.4 of the PSAR. The applicant noted in PSAR section 8.7.2, that the MCR will be further developed and the final MCR design will be provided in the future OL application for the NRC staff’s review as required by 10 CFR 50.34(f)(2)(iii). The applicant also stated in PSAR section 8.7.2 that it will perform the design-specific staffing analysis in the MCR and provide an exemption request to 10 CFR 50.54(m) as part of the future OL application.

The NRC staff reviewed the PSAR in accordance with 10 CFR 50.34(f)(2)(iv) and determined that the applicant stated that safety parameter displays for the MCR will be developed to the specifications required by 10 CFR 50.34(f)(2)(iv) along with specific guidance from NUREG-0711. The applicant also stated that the final design of safety parameter displays and the identification of critical safety functions and associated monitoring parameters will be further developed, validated, and described as appropriate in the future OL application.

The NRC staff reviewed the PSAR in accordance with 10 CFR 50.34(f)(2)(v) and determined that the applicant plans to design the MCR HSI in accordance with 10 CFR 50.34(f)(2)(v) using the guidance of RG 1.47. RG 1.47 provides supplemental guidance for implementing IEEE Std 603-2018 with respect to bypassed and inoperable status indication for plant safety systems. The applicant stated that the control room HSI design will be developed and validated prior to the future OL application.

The NRC staff reviewed the PSAR in accordance with 10 CFR 50.34(f)(2)(xi) and determined that 10 CFR 50.34(f)(2)(xi) is not applicable for the LMGS facility due to the design of the reactor primary cooling system. This regulation is among the list of non-applicable Three Mile Island requirements for the LMGS facility as discussed in SE section 1.4.

The NRC staff reviewed the PSAR in accordance with 10 CFR 50.34(f)(2)(xvii) and determined that the applicant's declaration that the LMGS MCR design includes instrumentation to monitor direct radiation levels and gaseous effluents at potential accident release points will comply with 10 CFR 50.34(f)(2)(xvii) is acceptable for the CP application. The NRC staff determined that because the LMGS facility does not have a traditional containment structure, in which the indications for containment pressure, water level, hydrogen concentration, and radiation intensity will not be available for MCR, other portions of 10 CFR 50.34(f)(2)(xvii) are not applicable to LMGS.

The NRC staff reviewed the PSAR in accordance with 10 CFR 50.34(f)(2)(xviii) in which the applicant stated that the HSI design and monitoring of inadequate core cooling will be developed as an indicator in the LMGS MCR. The primary parameters used for this indication are provided from a suitable combination of signals associated with the primary parameters used to monitor the critical safety function of "control heat removal." The completed HSI design for this indication will be described in the future OL application.

The NRC staff reviewed the PSAR in accordance with 10 CFR 50.34(f)(2)(xix) in which the applicant stated that the HSI for instrumentation adequate for monitoring plant conditions following licensing basis events will be developed in accordance with 10 CFR 50.34(f)(2)(xix) and the completed design for monitoring parameters will be provided in the future OL application.

The NRC staff reviewed the PSAR in accordance with 10 CFR 50.34(f)(2)(xxvi) and determined that 10 CFR 50.34(f)(2)(xxvi) is not applicable to LMGS because the Xe-100 design does not circulate radioactive materials outside of qualified barriers to radioactive release. This regulation is among the list of non-applicable regulations for the LMGS as discussed in SE section 1.4.

The NRC staff reviewed the PSAR in accordance with 10 CFR 50.34(f)(2)(xxvii), which states that the HSI design will include monitoring of in-plant radiation and airborne radioactivity under normal operating and licensing basis event conditions, as required by 10 CFR 50.34(f)(2)(xxvii).

The final HSI design and radiological monitoring systems and processes will be developed and described in the future OL application.

The NRC staff reviewed PSAR section 8.7.2 in accordance with 10 CFR 50.34(f)(3)(i), which states that an operating experience review (OER) implementation plan was developed as part of the overall HFE program and that the applicant used the guidance in NUREG-0711 to develop the methodology, process, and procedures. The final OER implementation plan will be included in the FSAR as part of the future OL application.

The NRC staff reviewed each of the above items in comparison to each of the regulations stated and concluded that the applicant provided sufficient information to satisfy the technically relevant requirements of 10 CFR 50.34(f)(2) and 10 CFR 50.34(f)(3) at the CP application stage for LMGS.

8.7.1.3 Development of the HFE Program

The NRC staff reviewed the description of the development of the HFE program in PSAR section 8.7.2, in which the applicant states that the HFE program is divided into 12 elements, consistent with the format used in NUREG-0711.

To verify that the applicant has established an HFE program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures in accordance with 10 CFR 50.34(f)(2)(ii), the NRC staff reviewed PSAR section 8.7.2. The applicant stated in PSAR section 8.7.2 that the LMGS HFE program plan will specifically adhere to the guidance found in NUREG-0711. The applicant also indicated that the LMGS HFE program plan will be developed under the LME QAP, which will include requirements for implementing procedures, processes, and forms. The applicant also discusses the establishment of the HFE team to develop the overall HFE program plan and to provide all program activities through the HFE issue tracking system. The applicant also indicated in PSAR section 8.7.2 that the final LMGS HFE program will be submitted prior to the OL application for the NRC staff's review.

The NRC staff reviewed the applicant's description of the LMGS HFE program plan and concluded that the applicant intends to incorporate the guidance found in NUREG-0711 as part of the development of the HFE program plan. The NRC staff finds that the development of the LMGS HFE program plan, as described by the applicant in the PSAR, will satisfy the requirements of 10 CFR 50.34(f)(2)(ii).

8.7.2 Conclusion

The NRC staff reviewed the applicable PDC 19 for the LMGS MCR and alternate location to shut down the reactor as required by 10 CFR 50.34(a)(3)(i) and the applicable regulations for the development of the HFE program plan as required by 10 CFR 50.34(f)(2) and 10 CFR 50.34(f)(3). The NRC staff finds that the applicant's development of the LMGS HFE program (including the OER plan) and design of the MCR are in accordance with 10 CFR 50.34(f)(2) and 10 CFR 50.34(f)(3) and are acceptable for the CP application stage. The NRC staff also finds that the adoption of PDC 19 for the LMGS MCR and alternate location to shut down the reactor is consistent with the requirements of 10 CFR 50.34(a)(3)(i). Based on its review, the NRC staff finds that the design of the LMGS MCR and development of the HFE program plan, as described in PSAR section 8.7, is consistent with the applicable regulatory guidance and meets

the regulatory requirements identified in 10 CFR 50.34(a)(3)(i),10 CFR 50.34(f)(2), and 10 CFR 50.34(f)(3).

8.8 Chemistry Program

In section 8.8 of its PSAR, LMGS provided a description of the chemistry program that will be established to monitor and control the chemistry of the helium coolant in the primary loop and water in the secondary loops to minimize corrosion of components and piping and stated that its chemistry program is an operational program, and details of the LMGS chemistry program will be provided in the OL application. The NRC staff determined that the description of the chemistry program is acceptable because it is an operational program that is not required for the CP and the applicant provided sufficient information regarding chemistry control needs and capabilities that are applicable to the preliminary LMGS design. The NRC staff agrees and notes that 10 CFR 50.34(b)(6)(ii) will require the OL application to include the managerial and administrative controls to assure safe operation, whereas 10 CFR 50.34(a)(7) does not require such information to be in the CP application. The NRC staff is not making any regulatory findings in this SE regarding the overall acceptability of the LMGS chemistry program because the application does not contain all the required information, that information is not required at the CP stage, and the applicant is not seeking to demonstrate compliance with 10 CFR 50.55a(f)(7) at this time.

8.9 Comprehensive Vibration Assessment Program (CVAP)

The NRC staff reviewed LME's comprehensive vibration assessment program (CVAP), which is used to identify, assess, and mitigate the potential adverse effects of flow, mechanically, and acoustically induced vibration on the reactor vessel internals (RVI). Per PSAR section 8.9, other SSCs comprising the primary helium circuit, which are identified as potential remote sources of vibration that could adversely affect the RVI, are also examined. The NRC staff assessed the adequacy of LMGS CVAP in support of the issuance of a CP.

The applicable regulatory requirements for the evaluation of LMGS CVAP for a CP are:

- 10 CFR 50.34(a)(3)(i); and
- 10 CFR 50.34(a)(3)(ii).

The applicable guidance for the evaluation of LMGS CVAP is:

- RG 1.20, Revision 4, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Startup Testing," (ML16056A338).

Note that although the scope of RG 1.20, Revision 4, does not explicitly cover advanced reactors like the Xe-100 design, it provides guidance for evaluating the design of nuclear power plant fluid systems to avoid adverse effects of vibration phenomena for a wide range of fluid systems, including Xe-100 systems, and is thus applicable to the review of LMGS CVAP for CP application.

The following PDCs, as defined in section 5.3 of the LMGS PSAR, apply to the review of LMGS CVAP:

- PDC 1, “Quality Standards and Records”; and
- PDC 4, “Environmental and Dynamic Effects Design Basis.”

8.9.1 Technical Evaluation

PSAR table 1.4-2 states that the LMGS CVAP fully conforms with RG 1.20, Revision 4. Accordingly, the NRC staff reviewed the PSAR against the guidance in RG 1.20.

RG 1.20, section C.1 requires that the applicant classify its reactor internal configurations as prototype, limited prototype, or non-prototype. In PSAR section 8.9.3, LME stated that the reactor internals of the first Xe-100 reactor module are classified as prototype. LME also stated that upon the successful implementation of the CVAP for the first XE-100 module, the subsequent three modules can be classified as non-prototype for purposes of vibration assessment. The NRC staff’s findings about the LMGS classification of reactor internals for the purpose of CVAP plan are summarized below:

- RG 1.20, section C.1.1 defines a “protoype” as a configuration of reactor internals that represents a first-of-a-kind or unique design because of the arrangement, design, size, or operating conditions. LMGS’s classification for the reactor internals of the first Xe-100 module as prototype is consistent with the guidance in RG 1.20 and therefore, is acceptable.

- RG 1.20, section C.1.1 states that after the NRC staff accepts the prototype design and the CVAP is completed with no adverse vibration or excessive loading, the prototype reactor internals are classified as a “valid prototype.” RG 1.20, section C.1.3 further states that reactor internals that have substantially the same arrangement, design, size, and operating conditions as a “valid prototype” can be classified as a “non-prototype.” Since the three subsequent LMGS reactor modules are expected to have similar arrangement, design, size, and operating conditions as the protype (i.e., the first LMGS reactor module unit), the NRC staff finds that the classification of non-prototype reactor internals of the three subsequent LMGS modules is consistent with the guidance in RG1.20, if the CVAP for the first LMGS module will be successfully completed and the NRC staff accepts the prototype design. In the review of the OL application, the NRC staff will confirm that the CVAP for the prototype reactor internals is successfully completed such that the reactor internals of the subsequent three modules at LMGS can be classified as non-prototype for the purpose of the vibration assessment.

The NRC staff reviewed PSAR sections 8.9.2 through 8.9.4, which describe the CVAP plan for the prototype reactor internals. The NRC staff finds that the LMGS CVAP plan for the prototype reactor internals is consistent with the guidance in RG 1.20, section C.2 and therefore, is acceptable because:

- Per PSAR section 8.9.2 and section 8.9.3, the CVAP plan contains the three major programs (i.e., vibration and stress analysis program, vibration measurement, and

inspection programs) required in RG 1.20, section C.2 to demonstrate the acceptable performance of reactor internals.

- PSAR section 8.9.2 defines the scope of CVAP program to include RVI components and other SSCs comprising the primary helium circuit that could be potential remote sources to induce adverse vibration of the RVI components. This definition of CVAP scope is consistent with the guidance in RG 1.20.

- The CVAP plan described in PSAR section 8.9.3 indicates that the CVAP plan will consider all the potential excitation mechanisms that could induce excessive vibration of the reactor internal components. This is consistent with the guidance in RG 1.20, section C.2. In addition, through auditing Reference 8.9-1, "Xe-100 Comprehensive Vibration Assessment Program," the NRC staff confirmed that all the potential excitation mechanisms identified in RG 1.20, section C.2 will be considered in the vibration and analysis program of the LMGS CVAP program.

- PSAR section 8.9.4 states that: "[t]he results obtained through CVAP execution will be documented in multiple stages in a CVAP Report, consistent with RG 1.20, section 2.4." Additionally, PSAR sections 8.9.4.2 through 8.9.4.4 state that the initial, preliminary, and final reports will fulfill the reporting guidance in RG 1.20, sections 2.4a for the initial report, 2.4b for the preliminary report, and 2.4c plus 2.4d for the final report. The NRC staff finds that the plan for the documentation of the prototype CVAP in PSAR section 8.9.4 complies with requirements a through d in RG 1.20, section C.2.

- PSAR section 8.9 provided the classification of the reactor internals and the scope of the CVAP. This is consistent with the guidance in RG 1.20, section C.2.5 for the information required for the CP application. Per RG 1.20, section C.2.5, the details of the CVAP, including the preoperational test procedures, power ascension program, and CVAP report can be submitted and reviewed later as part of the OL application process or pre-application for the OL application.

PSAR section 8.9.3 states that CVAP for the subsequent three modules will be consistent with RG 1.20, section C.4, which provides the requirements for non-prototype reactor internal CVAP. LMGS did not provide additional information for the CVAP of the subsequent three modules. Per table 1 of RG 1.20, CVAP for non-prototype reactor internals can be established based on the CVAP for the corresponding prototype reactor internals, with reduced scope compared to the CVAP for prototype reactor internals. Considering that LME provided sufficient information for CVAP of the prototype reactor internals in the first module in PSAR section 8.9, the NRC staff has adequate assurance that LMGS CVAP for the subsequent three modules will be consistent with RG 1.20 and will, if asked, provide feedback on the LMGS CVAP for the subsequent three modules in pre-application for the OL application and review it in the OL application process.

8.9.2 Conclusion

Based on the information provided in PSAR section 8.9, the NRC staff determined that the methods and procedures for developing the scope, approach, and stages of report for the RVI CVAP are consistent with RG 1.20. RG 1.20 provides an acceptable approach to quantitatively demonstrate that there were no adverse vibration phenomena or excessive loading to impact structural integrity of the reactor internals. Complying with RG 1.20 thus satisfies the

requirements in PDC 4 to protect safety-significant SSCs from dynamic effects. RG 1.20 also provides guidance to appropriately document the CVAP plan and results. By complying with RG 1.20, section C.2.4, the LMGS CVAP plan satisfies the requirements of PDC 1 to maintain appropriate records of the design and testing of safety-significant SSCs. By supporting requirements in PDCs 1 and 4, the LMGS CVAP in PSAR section 8.9 satisfies the requirements in 10 CFR 50.34 (a)(3)(i) and (a)(3)(ii). Therefore, the NRC staff concludes that the RVI CVAP program as described in the PSAR, provides reasonable assurance that the applicable regulatory requirements and guidance for the issuance of a CP are met and additional details of the CVAP can be left for consideration in the FSAR.

8.10 Reliability, Availability, and Maintainability Program

The NRC staff reviewed PSAR section 8.10, which provides a high-level description of the reliability, availability, and maintenance program (RAM). The PSAR states the objectives of the program are to define, evaluate, and implement strategies to ensure that reliability, availability, and maintainability of passive SSCs that perform functions requiring special treatment for DID adequacy.

The NRC staff notes that this program, once fully developed in the OL application, would be needed to support a regulatory finding under 10 CFR 50.34(b)(6)(iv), which is applicable at the OL application stage. Because the applicant has not provided all of the required information on the RAM in its CP application and is not seeking to demonstrate compliance with this regulation at the CP application stage, the NRC staff is not making any regulatory findings pertaining to this section. The NRC staff reviewed PSAR section 8.10.2, which discussed the RAM DMA and finds that using XI.2 Article VII-3 in addition to the materials qualification methodology described in PSAR section 3.8 is a reasonable approach. The NRC staff will evaluate the RAM 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 (PSAR section 8.2), which includes passive SSCs that perform functions requiring special treatment for DID adequacy.

9 CONTROL OF ROUTINE PLANT RADIOACTIVE EFFLUENTS, LIQUID RADWASTE, SOLID RADWASTE, AND PLANT CONTAMINATION

This chapter of the safety evaluation (SE) summarizes the U.S. Nuclear Regulatory Commission (NRC or Commission) staff's review and evaluation of the Long Mott Generating Station (LMGS) preliminary safety analysis report (PSAR) chapter 9, "Control of Routine Plant Radioactive Effluents, Liquid Radwaste, Solid Radwaste, and Plant Contamination," which contains preliminary descriptions of the release of liquid and gaseous effluents, the liquid and gaseous waste treatment systems, plans to minimize plant contamination, and the control of solid radioactive waste.

The applicable regulatory requirements for the evaluation of the LMGS PSAR chapter 9 are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection Against Radiation";
- 10 CFR 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents – nuclear power reactors";
- 10 CFR 50.40, "Common standards";
- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents"; and
- Title 40 of the *Code of Federal Regulations* (40 CFR) Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations."

The applicable guidance for the evaluation of the LMGS PSAR chapter 9 is as follows:

- Division of Advanced Reactors and Non-Power Production and Utilization Facilities (DANU) Interim Staff Guidance (ISG) DANU-ISG-2022-03, "Advanced Reactor Content of Application Project, Chapter 9, 'Control of Routine Plant Radioactive Effluents, Plant Contamination and Solid Waste'" (ML23277A141);
- Regulatory Guide (RG) 1.21, Revision 3, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste" (ML21139A224);
- RG 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I" (ML003740384);
- RG 1.143, Revision 2, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants" (ML013100305); and

- RG 4.21, Revision 0, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning” (ML080500187).

The following principal design criteria (PDC), as defined in PSAR section 5.3, apply to the evaluation of the LMGS PSAR chapter 9:

- PDC 60, “Control of Releases of Radioactive Materials to the Environment”;
- PDC 61, “Fuel Storage and Handling and Radioactivity Control”;
- PDC 63, “Monitoring Fuel and Waste Storage”; and
- PDC 64, “Monitoring Radioactive Releases.”

9.1 Gaseous Effluents

PSAR section 9.1 describes the structures, systems, and components (SSCs) associated with the control of gaseous effluents. During normal operations of LMGS, the primary sources of gaseous radiological effluent are from: (1) tritium production and evaporation from the reactor cavity cooling system (RCCS), and (2) a release fraction of the steady-state radionuclide inside the helium pressure boundary (HPB). The steady-state inventory in the HPB is represented by circulating gas and vapor adsorption on HPB surfaces and on accumulated graphite and metal dust particles, decay, and removal via the helium purification system (HPS).

Gaseous effluent release points include the reactor building (RB) shield structure (SST) heating, ventilation, and air conditioning (HVAC); the consolidated discharge point for the helium storage facility (HSF), which includes the fuel handling annex building, inter unit access tunnel, and canister processing facility; and the radwaste building HVAC.

PSAR section 7.3.7 describes the HPB and PSAR section 7.3.10 describes the nuclear island HVAC, which controls gaseous effluent releases.

9.1.1 Technical Evaluation

As discussed in DANU-ISG-2022-03, the NRC staff reviews gaseous effluents to ensure that the construction permit (CP) application provides assurance that the limits on the release of radioactive liquid and gaseous effluents will meet the relevant requirements and to ensure that an application provides information on the design of the gaseous radwaste management system.

9.1.1.1 Gaseous Radwaste System Descriptions

The NRC staff reviewed the LMGS PSAR to determine whether, as required by 10 CFR 50.34a(b)(1), the CP applicant included the preliminary design of equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations in the PSAR. The NRC staff also reviewed the LMGS PSAR to ensure that the preliminary design meets PDC 60 and PDC 61. PDC 60 specifies that the nuclear power unit shall be designed to control the suitably of the release of radioactive materials in gaseous and liquid effluents. In addition, PDC 60 requires that the design shall

include system descriptions that demonstrate that the waste systems have sufficient holdup capacity for retention of gaseous and liquid effluents containing radioactive materials. PDC 61 requires that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions and have suitable shielding for radiation protection. For the CP review, the NRC staff reviewed the system descriptions for: the helium service system (HSS); the HPS (which is a subsystem of the HSS); and the nuclear island heating, ventilation, and air conditioning (NIHV) in PSAR sections 7.3.7, 7.3.10, 9.1.3, 9.1.4, 9.1.5, and 9.1.6 to determine whether the system descriptions in the PSAR meet 10 CFR 50.34a(b)(1) and PDC 60. In addition, the NRC staff ensured that high-level descriptions of shielding were provided in the PSAR. The NRC staff would evaluate the adequacy of shielding to meet PDC 61 as part of the review of an operating license (OL) application.

The NRC staff reviewed PSAR section 9.1.3, which states that there are three monitored release points from the NIHV. Gaseous effluent release points include the RB SST HVAC; the consolidated discharge point for the HSF, which includes the fuel handling annex building, inter unit access tunnel, and canister processing facility; and the radwaste building HVAC.

The NRC staff reviewed PSAR sections 7.3.7 and 9.1.4, which provide information on the HSS and describe the purpose of the HSS and the HPS subsystem to purify the helium coolant. PSAR section 7.3.7 also provides a description of the equipment available to purify helium. In addition to the HPS, the gas analysis and sampling system (GASS) is a subsystem of the HSS that includes detectors, monitors, and a sampling station which are used to assess the overall helium purity, quantify contaminants, and quantify radioactivity at various stages of the process flow. PSAR section 7.3.7.2 describes the flow path of helium as it passes through the HPS components, provides information on the individual components, and details sources of liquid that are directed to the nuclear island liquid radwaste system (NILR). PSAR figure 7.3.7-1, provides the flow diagram that describes the functional arrangement of the HPS.

The NRC staff reviewed PSAR sections 7.3.10 and 9.1.5, which provide the purposes of the NIHV to control airborne contamination below safe working levels, filter and monitor the air supply, maintain directional flow for contamination control, clean exhaust air before discharge to atmosphere, and monitor contaminants in working areas and release to the environment. Under normal operating conditions, the exhaust of the HVAC is monitored, and additional filtration is added when high radioactivity is detected. PSAR section 9.1.5 states that limits on routine releases of gaseous waste will be established in an OL application using the methodology described in RG 1.109. PSAR figure 7.3.10-1 provides a diagram of the buildings and corresponding systems and release points serviced by each of the three NIHV release points.

The NRC staff reviewed PSAR section 9.1.6, which provides information on the codes and standards applicable to the design of the HSS and NIHV. PSAR table 9.1-9 provides a list of codes and standards applicable to the NIHV. In addition, PSAR section 9.1.6 states that the HSS and NIHV design has suitable shielding for radiation protection, residual heat removal, and appropriate containment, confinement, and filtering systems to assure adequate safety.

The NRC staff determined that the PSAR sections above provide a description of preliminary design information for systems responsible for the control of gaseous effluents. In addition, the NRC staff conducted an audit (ML26293A340) to confirm the extent of the preliminary design of the HSS and NIHV to control gaseous waste. During the audit, the NRC staff reviewed design documents to better understand the configuration of the HSS and NIHV. Consistent with the

requirements specified in 10 CFR 50.34a(b)(1), the NRC staff determined that the PSAR sections summarized above include a description of the preliminary design of equipment in the HSS and NIHV that will maintain control over radioactive materials in gaseous effluents produced during normal reactor operations. The NRC staff determined that these descriptions meet the PDC 60 requirements for control of gaseous effluent because the PSAR contains descriptions of the HPS and NIHV systems used to control gaseous effluent releases and the descriptions provide that these gaseous effluent releases will be monitored and will impose additional filtration on high radiation signals in the NIHV to ensure that releases are below the 10 CFR Part 20, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," Table 2 effluent concentration limits. In addressing PDC 61, the NRC staff determined that PSAR section 9.1 stated that Long Mott Energy, LLC (LME) will evaluate shielding for radiation protection in the OL application, and LME provided additional design details in the audit such as identifying areas of the HPS that will require radiation shielding. The NRC staff determined that this is acceptable because LME is not required to address shielding for radiation protection during the CP stage and the NRC staff will review the details of radiation shielding during the review of any OL application.

Based on the review described above, the NRC staff has determined that the PSAR adequately describes the design of the HSS (which includes the HPS) and NIHV for the control of gaseous waste and provides the information needed to show compliance with 10 CFR 50.34a(b)(1), PDC 60, and PDC 61 for the CP application stage of the review. The NRC staff will review the final HSS and NIHV designs provided by LME to support any OL application to ensure compliance with 10 CFR 50.34a(c), PDC 60, and PDC 61.

9.1.1.2 Gaseous Effluent Source Terms

The NRC staff reviewed the LMGS PSAR to determine that, as required by 10 CFR 50.34a(b)(2)(ii), the CP applicant specified an estimate of the quantity of each of the principal radionuclides in gaseous waste that are expected to be released annually to unrestricted areas produced during normal reactor operations. The estimated quantity of radioactive materials is also referred to as the source term.

The NRC staff reviewed PSAR sections 9.1.1 and 9.1.2, as well as PSAR tables 9.1-1 and 9.1-2, to confirm that the applicant has specified an estimate of the quantity of radionuclides in gaseous waste that are expected to be released annually for compliance with 10 CFR 50.34a(b)(2)(ii). PSAR section 9.1.1 provides information on the gaseous effluent source terms. During normal operations of the LMGS, the primary sources of gaseous radiological effluent are tritium production and evaporation from the RCCS as well as a release fraction of the steady-state radionuclide inside the HPB. PSAR section 9.1.2 states that the RCCS tritium concentration, RCCS evaporation rate, and the steady-state radionuclide inventory are used to determine the source term. PSAR table 9.1-1 provides additional information on the preliminary radionuclide inventory estimate of the HPB. In addition, PSAR table 9.1-2 provides the annual estimated gaseous effluent releases from all modules.

In addition, the NRC staff audited the details of how PSAR tables 9.1-1 and 9.1-2 were determined. LME indicated that the preliminary estimates are based on an assumed release rate of the coolant inventory. The NRC staff's review of the source term information provided in PSAR tables 9.1-1 and 9.2-2, and the understanding gained by the NRC staff during the audit, allowed the NRC staff to determine that the estimated source term information is appropriate

because it considers the principal radionuclides expected from operations and is reflective of the reactor design information available during this CP application review.

Based on the above, the NRC staff has determined that the PSAR adequately describes the annual gaseous effluent releases consistent with the requirements of 10 CFR 50.34a(b)(2)(ii) for a CP application. The NRC staff would also review the effluent releases at the OL stage to ensure compliance with 10 CFR 50.34a(c)(2), which calls for a revised estimate of the information provided during the CP under 10 CFR 50.34a(b)(2)(ii).

9.1.1.3 Dose Analysis for Gaseous Effluent Releases

The NRC staff reviewed the LMGS PSAR to determine whether, as required by 10 CFR 50.34a(a), the CP applicant identified the design objectives for equipment to control releases of radioactive materials in effluents at nuclear power reactors, including a description of design objectives for keeping levels of radioactive material in effluents to unrestricted areas as low as is reasonably achievable (ALARA). The requirements in 10 CFR 20.1301, “Dose limits for individual members of the public,” specify the 1 mSv (100 mrem) annual public dose limit. In addition, 10 CFR 20.1301(e) specifies that licensees shall also comply with the standards in 40 CFR Part 190. The requirements in 10 CFR 20.1302(b)(1) specify that a licensee shall show compliance with 10 CFR 20.1301 by demonstrating through measurement or calculation that the total effective dose equivalent to the individual likely to receive the highest dose from operation does not exceed the 1 mSv (100 mrem) annual dose limit.

The NRC staff reviewed PSAR section 9.1.2, which states that the gaseous effluent releases are assumed to be continuous releases at the site boundary, the site exclusion area boundary, the RB SST, the HSF, and the fuel handling annex building (FHAB). In addition, PSAR section 9.1.2 states that the population dose and individual member of the public dose is calculated using the GASPAR II model of the NRCDose computer code. For the dose analysis performed, members of the public are not assumed to occupy the site boundary continuously whereas the nearest resident is assumed to occupy the location continuously. At the site boundary, only gaseous effluent releases and direct radiation from the spent fuel intermediate storage facility are assumed to contribute to offsite doses. To address the design objectives for keeping levels of radioactive material in effluents to unrestricted areas ALARA, the applicant states in PSAR chapter 9 that the chapter utilizes the numerical guidance contained in 10 CFR Part 50, Appendix I, to define design objectives for equipment to control releases of radioactive material in effluents ALARA.

The NRC staff reviewed PSAR table 9.1-2, which provides the preliminary annual effluent release rate. PSAR table 9.1-3 and PSAR table 9.1-6 provide the population dose estimates considering the annual effluent release rates. PSAR table 9.1-4 provides the dose analysis for the nearest resident located 1400 meters (0.87 miles) from the site. PSAR table 9.1-5 provides the dose analysis at the site boundary located 400 meters (0.25 miles) from the outer walls of the RB/FHAB/HSF. PSAR table 9.1-7 and PSAR table 9.1-8 provide comparisons of preliminary annual effluent dose estimates to 10 CFR 20.1301 dose limits and 40 CFR Part 190 dose limits, respectively.

The NRC staff performed a confirmatory calculation using the GASPAR II model of the NRCDose computer program using information provided in the PSAR and additional information provided during the audit. The NRC staff’s confirmatory analysis showed results consistent with those specified by the applicant in the PSAR tables 9.1-7, 9.1-8, and 9.1-10.

The NRC staff confirmed the results of LME's assessment for releases from the site through confirmatory calculations using the GASPAR II computer code. The NRC staff determined that these results are acceptable because the use of the GASPAR II model is consistent with RG 1.109, which provides an acceptable method for determining the doses from gaseous effluent releases. The results of the dose analysis show that gaseous releases are below the dose objectives in 10 CFR Part 50, Appendix I, and the calculated annual doses demonstrate that the doses will not exceed the public dose limits provided in 10 CFR 20.1301 and 40 CFR Part 190.

The NRC staff determined that preliminary information about the effluent sources and dose analysis on the facility's proposed effluent releases meet the CP requirements for 10 CFR 50.34a(a) because proposed preliminary effluent releases from the LMGS facility are below the dose objectives specified in 10 CFR Part 50, Appendix I; 10 CFR 20.1301; and 40 CFR Part 190. The NRC staff will review final effluent releases at the OL stage to ensure that the requirements in 10 CFR 20.1301, 40 CFR Part 190, 10 CFR 50.34a(c) and the 10 CFR Part 50, Appendix I dose criteria are met.

9.1.2 Conclusion

Based on its evaluation above, the NRC staff concludes that the information in PSAR section 9.1 is sufficient and meets the applicable guidance and regulatory requirements identified in this section for the issuance of a CP in accordance with 10 CFR 50.34a(a), 10 CFR 50.34a(b)(1), 10 CFR 50.34a(b)(2), and PDCs 60 and 61.

9.2 Liquid Radwaste

PSAR section 9.2 describes the SSCs associated with the control of liquid radwaste (LRW). During normal operations of the LMGS, the primary sources of LRW are generated by the RB, FHAB, HSF, and the radwaste building (RWB). These buildings may contain leakage from fluid systems that are collected by the NILR and transferred to the liquid radwaste handling system (LRWH). The LRW generated at LMGS is handled by the LRWH. The LRWH is designed to provide means to collect, handle, and process potentially low-activity waste that originates from nuclear island (NI) buildings at the LMGS site. The LRWH is classified as non-safety related with no special treatment (NST).

9.2.1 Technical Evaluation

As discussed in DANU-ISG-2022-03, the purpose of the NRC staff's review of LRW is to ensure that a CP application provides assurance that the limits on the release of radioactive liquid and gaseous effluents will meet the relevant requirements and to ensure that an application provides information on the design of the LRW management system.

9.2.1.1 Liquid Radwaste System Description

The NRC staff reviewed the LMGS PSAR to determine whether, as required by 10 CFR 50.34a(a), the CP applicant included a description of the preliminary design of equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations. The NRC staff also reviewed the LMGS PSAR to ensure that the preliminary design meets PDC 60. PDC 60 specifies that the nuclear power unit shall be designed to control the suitably of the release of radioactive materials in gaseous and liquid

effluents as well as the ability to handle radioactive solid wastes produced during normal reactor operation. In addition, PDC 60 requires that the design shall include system descriptions that demonstrate that the waste systems have sufficient holdup capacity for retention of gaseous and liquid effluents containing radioactive materials.

The NRC staff reviewed the information in PSAR section 9.2 to verify that the LRWH is adequately described and has demonstrated sufficient holdup capacity for liquid waste. The NRC staff reviewed PSAR section 1.1.4.3.7, which states that LRW is collected by the NILR and transferred to the RWB for treatment, and is monitored prior to discharge via the LRWH. PSAR section 9.2.3, states that the LRWH is designed to collect, handle, and process potentially low-activity liquid waste that originates from NI buildings that have the potential to contain radioactivity or contamination. LRW is collected via floor drains or equipment drains and is transferred by gravity, draining into designated NILR sump pits. From the sump pits, the LRW is pumped to the LRWH system in the RWB. The LRWH system also provides for the handling and transfer of spent ion-exchange resin and higher activity liquid waste via a dedicated pump to the solid radwaste handling system (SRWH). PSAR section 9.2.2 states that there are no plans for measurable onsite releases of liquid effluent from the LMGS.

The NRC staff reviewed PSAR table 9.2-1, which provides the preliminary estimates of LRW generation, and PSAR table 9.2-3, which provides the available estimated volumes within the sumps and tanks that may contain LRW. PSAR figure 9.2-1 provides the functional arrangement of the LRWH.

For the description of the components of the LRWH required by 10 CFR 50.34a(a), the NRC staff has determined that the applicant adequately described the LRWH in PSAR sections 9.2.2 and 9.2.3. The information contained in PSAR figure 9.2-,1 provided the NRC staff with insights on the high-level functional arrangement of the LRWH to understand the LRWH interfaces with the NILR, various sampling stations, potential temporary equipment, the SRWH, and disposal connections. In addition, the NRC staff reviewed the information contained in PSAR tables 9.2-1 and 9.2-3 to understand the available capacity to store LRW. Based on the total volume/day/module generated and available tank volumes provided in these PSAR tables, the NRC staff determined that LME has roughly 85 days of storage capacity, which increases to approximately 106 days of capacity when considering the sump volumes available.

During the audit associated with the LMGS CP application, the NRC staff requested details related to the connections of the NILR and LRWH to confirm the LRWH's ability to process and control LRW in accordance with 10 CFR 50.34a(a) and PDC 60. During the audit, the NRC staff held discussions with LME to understand how the NILR and LRWH systems interface with each other and to understand the stated capacities for LRW storage. The audit discussion clarified the NRC staff's understanding of the differences between the NILR and the LRWH and further confirmed the NILR's purpose to collect LRW from buildings that are expected to produce LRW and then direct the LRW to the LRWH, which is consistent with the description provided in PSAR sections 1.1.4.3.7 and 9.2.3. Audit discussions also confirmed the available LRW storage information provided in PSAR tables 9.2-1 and 9.2-3. Using the information provided in PSAR tables 9.2-1 and 9.2-3, and applying engineering judgment, the NRC staff determined that LMGS has sufficient LRW storage.

Based on the information and analysis above, the NRC staff determined that the LMGS design will safely store and process liquid waste from all interfaces to the LRWH because the receiving

tanks and waste monitoring tanks have sufficient capacity to store LRW from plant operations as required by PDC 60.

Based on the evaluation described above, the NRC staff has determined that the PSAR adequately describes the design of the LRWH and provides the information needed to show compliance with 10 CFR 50.34a(a) and PDC 60 for the CP application stage of the review. The NRC staff will review the final LRWH design provided by LME to support any OL application to ensure compliance with 10 CFR 50.34a(c) and PDC 60.

9.2.1.2 Liquid Radwaste Equipment Design Requirements

The NRC staff reviewed the LMGS PSAR to ensure that the preliminary design information meets PDC 61. PDC 61 requires that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions. These systems shall be designed: (1) with suitable shielding for radiation protection, (2) with appropriate containment, confinement, and/or filtering systems, (3) with sufficient residual heat removal, and (4) to prevent significant reduction in fuel storage coolant inventory during accident conditions including operational occurrences, design basis events, and design basis accidents, as appropriate. For the LMGS CP review, the NRC staff reviewed the LRWH against the design guidance in RG 1.143 and confirmed that high-level descriptions of shielding were provided. In addition, the NRC staff evaluated if additional design features were necessary to retain LRW using the supporting analysis in PSAR section 2.4.13. The NRC staff reviewed PSAR section 9.2.4, which states that the LRWH is an NST system. This PSAR section states that the LRWH uses engineering controls and procedures to achieve doses to the public ALARA. PSAR table 9.2-4 provides the codes and standards applicable to the design of the LRWH.

The NRC staff reviewed PSAR table 1.4-2 and PSAR section 9.2.4, which states that LMGS is in partial conformance with RG 1.143 and that the LRWH system is in conformance with regulatory position section C.1 of RG 1.143. In addition, PSAR table 1.4-2 states that the methodologies described in PSAR section 5.1 and section 6.1.3 replace the methodologies of RG 1.143 regulatory positions C.5 and C.6, respectively. The NRC staff evaluated these methodologies in SE sections 5.1 and 6.1. For systems related to LRW and solid radwaste, PSAR table 1.4-2 demonstrates that LMGS will generally conform to regulatory positions C.1, C.2, and C.3; LMGS will conform to the codes and standards listed in table 1 of RG 1.143, where applicable; and deviations/exceptions will be stated in the final safety analysis report (FSAR) if identified in any OL application.

The NRC staff reviewed PSAR section 9.2.4, which states that the LRWH includes process filtering systems and that the storage tanks provide suitable shielding where appropriate. The PSAR further states that spent resin storage tanks are provided with an enclosure that satisfies shielding requirements. The NRC staff's review of the LRWH in support of the design requirements for PDC 61 finds that additional details regarding radiation shielding and component source terms will need to be evaluated at the OL stage when those details are provided. The NRC staff determined that this is acceptable because LME is not required to address shielding for radiation protection during the CP stage and the NRC staff will review the details of radiation shielding during the review of any OL application. The NRC staff also understands that the design of the LRWH will apply ALARA design principles which will be fully evaluated as part of the review of any OL application.

To ensure adequate safety under normal and postulated accident conditions for PDC 61, the NRC staff reviewed PSAR section 2.4.13, which provides the LMGS dose estimates resulting from the accidental release of LRW. Within this PSAR section, LME states that the release of LRW results in negligible doses when taking into consideration the attenuation factors described in PSAR section 2.4.13 and PSAR tables 2.4.13-3 and 2.4.13-4. The NRC staff's evaluation of the transport of radionuclides is in SE section 2.4.13. Based on the results provided in the PSAR, if the resulting doses from the transport of radionuclides maintain releases below the public dose limit of 100 mrem per year as specified in 10 CFR 20.1301, this would support an NRC staff finding at the OL stage that the design does not need additional mitigating features to prevent or minimize public exposure in the event of an accidental release of LRW. In its review of any OL application, the NRC staff will confirm that any accidental releases are maintained below dose limits and verify that no additional mitigative design features are needed to control releases.

The NRC staff has determined that the PSAR adequately describes the design of the LRWH and that the PSAR provides the information needed to show compliance with PDC 61 for the CP application. The NRC staff has determined that the references made in PSAR table 9.2-4, are acceptable because the design of the LRWH system is consistent with the guidance in RG 1.143 and because LMGS will provide the final LRWH design for the NRC staff's review during the OL stage, consistent with 10 CFR 50.34a(c). The NRC staff also determined that RG 1.143 is the appropriate design guidance because it is consistent with design guidance previously used by light-water reactors (LWRs) and that the reference made to the methodologies in PSAR sections 5.1 and 6.1.3 are appropriate given LME's safety classification methodology and the classification of the LRWH as an NST system. The NRC staff's evaluation of the safety classification methodology described is documented in SE section 5.1. The NRC staff's review of PSAR section 2.4.13 also determined that the expected accidental tank releases results in a negligible dose, and therefore no additional design features are necessary to maintain doses less than the public dose limits in 10 CFR 20.1301. The NRC staff will review the final LRWH design at the OL stage to ensure compliance with 10 CFR 50.34a(c), PDC 60, and PDC 61.

9.2.1.3 Liquid Radwaste Source Terms

The NRC staff reviewed the LMGS PSAR to determine whether, as required by 10 CFR 50.34a(b)(2), the CP applicant specified an estimate of the quantity of radionuclides in both gaseous and liquid waste that are expected to be released annually.

The NRC staff reviewed PSAR section 9.2.1, which provides information on the LRW source terms. This PSAR section states that LRW is generated from leakage of fluid systems in the radiologically controlled area (RCA), liquid wastes collected from floor drains in the RCA, LRW generated from the steam generator (SG) following a SG tube rupture, LRW generated during decontamination of equipment, tritium liquid waste generated from the HPS and RCCS processing water, and leakage from sampling or sample analysis activities. PSAR section 9.2.2 states that the applicant plans for no measurable onsite LRW discharge to the environment, with all LRW to be shipped offsite. Tritium is removed from the helium thermal transfer fluid as described by the HPS processes in PSAR section 7.3.7.2 and directed to the LRWH system. LRW is stored within the RWB and is periodically collected and disposed of by third party radioactive waste management vendors. PSAR section 9.2.2 also states that residual tritium in liquid waste will be small based on preliminary estimates and that a more detailed assessment will be provided to support any OL application.

The NRC staff reviewed PSAR table 9.2-2, which provides the estimated annual activity for LRW generated at LMGS. A footnote to PSAR table 9.2-2 addresses the preliminary LRW source term estimates by indicating that LRW activity estimates are based on the LRW subsystem design information from U.S. Department of Energy (DOE) high-temperature gas-cooled reactor (HTGR) DOE-HTGR-87-021. The PSAR states that DOE-HTGR-87-021 "is considered the best estimate of the liquid radwaste sources for the initial representation for the liquid radwaste activity of the Xe-100 design." The footnote additionally addresses the applicability of DOE-HTGR-87-021 to the Xe-100 design. The NRC staff's review of DOE-HTGR-87-021 finds that this document reflects the design of a LRWH system for a modular high-temperature gas reactor design which provides estimates for LRW. In addition, the NRC staff determined that DOE-HTGR-87-021 also describes sources of the LRW that are appropriate for use in providing preliminary estimates of LRW for the Xe-100 design because of the similarity in design, LRW sources, and power level. During the audit, the NRC staff reviewed information in the DOE-HTGR-87-021 document and confirmed that this document provided LRW activities and volumetric estimates that are presented in PSAR tables 9.2-1 and 9.2-2. The NRC staff's review of the information provided to support PSAR table 9.2-2 confirmed that the preliminary estimate for liquid source terms is determined from the information in DOE-HTGR-87-021. The NRC staff further verified, through the audit and confirmatory calculations, how the liquid source term information contained in PSAR table 9.2-2 was determined.

The NRC staff determined that the PSAR adequately describes the preliminary information on the LRW consistent with the requirements in 10 CFR 50.34a(a) for a CP application because LMGS has provided a preliminary estimate for annual LRW activity that is based on LRW estimates from a similar reactor design which produces LRW in the same ways anticipated for the LMGS facility. The NRC staff will review the final LRW sources at the OL stage to ensure compliance with the 10 CFR 50.34a(c)(2).

9.2.2 Conclusion

Based on its evaluation above, the NRC staff concludes that the information in the LMGS PSAR section 9.2 is sufficient and meets the applicable guidance and regulatory requirements identified in this chapter for the issuance of a CP in accordance with 10 CFR 50.34a(a), 10 CFR 50.34a(b)(2), and PDCs 60 and 61.

9.3 Solid Waste Management

PSAR section 9.3 describes the SRWH that is primarily located within the RWB. The SRWH is designed to process both wet and dry solid waste resulting from normal plant operation. The dry active waste source term estimates are based on graphite dust that is captured directly on filters, or contamination on equipment or other surfaces. The major components of the SRWH are the drum filling station, sorting table, compactor, drum capping station, and drum assay measuring station.

9.3.1 Technical Evaluation

As discussed in DANU-ISG-2022-03, the purpose of the NRC staff's review of solid radwaste processing is to ensure that an application provides information on the design of the solid radwaste management system.

9.3.1.1 Solid Radwaste System Description

The NRC staff reviewed the LMGS PSAR to determine whether, as required by 10 CFR 50.34a(a), the CP applicant included a general description of the provisions for packaging, storage, and shipment offsite of solid waste containing radioactive materials resulting from the treatment of gaseous and liquid effluents and from other sources. The NRC staff also reviewed the LMGS PSAR to ensure that the preliminary design meets PDC 60. PDC 60 specifies that the nuclear power unit systems, which may contain radioactivity, shall be designed to assure adequate safety under normal and postulated accident conditions. In addition, PDC 60 specifies that system descriptions shall demonstrate that the waste systems have sufficient holdup capacity for the SRWH to control the release of gaseous and liquid effluents containing radioactive materials.

The NRC staff reviewed PSAR section 9.3.3 and Environmental Report (ER) section 3.5.2, which provide an overview of the SRWH. PSAR section 9.3.3 states that the SRWH is primarily located within the RWB and that the SRWH is designed to process both wet and dry solid waste resulting from normal plant operation. The SRWH includes an interim storage area in the RWB for packaged dry and wet solid radwaste. PSAR section 9.3.3 further states that filled drums are stored on site until they are shipped offsite for disposal. In addition, this PSAR section states that offsite disposal transport will be in accordance with 10 CFR Part 71, “Packaging and Transportation of Radioactive Material,” which will be addressed by the process control program (PCP). PSAR figure 9.3-1 provides the functional arrangement of the SRWH. ER section 3.5.2.3 additionally states that low-level radwaste shipments will be transported to the Texas Compact Waste Facility operated by Waste Control Specialists, LLC., (WCS) in Andrews, Texas.

The NRC staff reviewed the information in PSAR section 9.3.3 to confirm that the SRWH is adequately described and that the SRWH design has sufficient holdup capacity for solid waste in support of PDC 60. The NRC staff determined that LME demonstrated sufficient holdup capacity because the PSAR provides specific plans to store and ship waste offsite. The NRC staff also determined that LME has adequately described the SRWH and storage area locations in PSAR section 9.3.3 and ER section 3.5.2 because the applicant provides sufficient high-level descriptions of waste volumes generated and describes the designated storage area within the RWB for storage of solid waste. The information contained in figure 9.3-1 also allowed the NRC staff to understand the high-level functional arrangement of the SRWH and interfaces between components of the SRWH system.

The NRC staff has determined that LME has described the SRWH by providing details on the methods for the packaging, storing, and shipment of waste containing radioactive materials in accordance with 10 CFR 50.34a(a). In support of the PDC 60 requirement to assure adequate safety under normal and postulated accident conditions, the NRC staff determined that LME has adequately described that the SRWH will be designed to collect, segregate, process, sample, and monitor waste generated from normal operations. Furthermore, the NRC staff confirmed that the SRWH will control and confine radionuclides against uncontrolled releases and employ design features as discussed in PSAR section 9.2 to control radioactive releases.

The NRC staff has determined that the PSAR adequately describes the design of the SRWH and provides the information needed to show compliance with 10 CFR 50.34a(a) and PDC 60, for the CP application. The NRC staff will review the final SRWH design at the OL stage to ensure compliance with 10 CFR 50.34a(c) and PDC 60.

9.3.1.2 Solid Radwaste Equipment Design Requirements

As specified by PDC 61, the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity must be designed to ensure adequate safety under normal and postulated accident conditions. These systems shall be designed: (1) with suitable shielding for radiation protection, (2) with appropriate containment, confinement, and/or filtering systems, (3) with sufficient residual heat removal, and (4) to prevent significant reduction in fuel storage coolant inventory during accident conditions including operational occurrences, design basis events, and design basis accidents, as appropriate. For the CP review, the NRC staff reviewed the SRWH in accordance with the design guidance in RG 1.143 and ensured that high-level descriptions of shielding were provided.

The NRC staff reviewed PSAR table 1.4-2 and PSAR section 9.3.4, which state that LMGS is in partial conformance with RG 1.143, and that the SRWH is in conformance with regulatory position section C.3 of RG 1.143. In addition, PSAR table 1.4-2 states that the methodologies described in PSAR section 5.1 and PSAR section 6.1.3 replace the methodologies of RG 1.143 regulatory positions C.5 and C.6, respectively. The NRC staff evaluated these methodologies in SE sections 5.1 and 6.1. For systems related to LRW and solid radwaste, the PSAR table indicated that LMGS will generally conform to regulatory positions C.1, C.2, and C.3; LMGS will conform to the codes and standards listed in table 1 of RG 1.143, where applicable; and deviations/exceptions will be stated in the FSAR, if identified in any OL application. The NRC staff determined that while RG 1.143 was developed for LWRs, the RG provides appropriate design guidance for the SRWH because similar systems to those contemplated in the RG will be used by LMGS for waste treatment based on the information provided in the CP application.

The NRC staff reviewed PSAR section 9.3.4, which states that the SRWH is designed to maintain occupational dose ALARA. The SRWH will be in a shielded area and is stated to meet the radiation zoning criteria established for anticipated occupancy. Design features included as part of the SRWH to ensure occupational dose ALARA are stated to be remotely operated valves, shielding of components containing radioactive material, and operational controls related to the access of areas containing radioactive materials.

In its review of the SRWH in support of the design requirements for PDC 61, the NRC staff noted that additional details regarding radiation shielding and component source terms will be provided in support of any OL application for the NRC staff's review and evaluation. The NRC staff also understands that the design of the SRWH will apply ALARA design principles which will be fully evaluated as part of the review of any OL application.

The NRC staff has determined that the PSAR adequately describes the preliminary design of the SRWH system and provides the information needed to show compliance with PDC 61 for the CP application review because LMGS applies the design guidance in RG 1.143, as indicated above, with exceptions to regulatory positions C.5 and C.6, and describes methods to maintain occupational doses ALARA. The NRC staff will review the final SRWH design at the OL stage to ensure compliance with PDC 61.

9.3.1.3 Operational Controls and Process Control Program

The NRC staff reviewed PSAR section 9.3.3, which describes that offsite disposal and transport will be in accordance with 10 CFR Part 71, with final modality confirmed as part of the development of a site-specific waste management PCP. The NRC staff notes that the applicant

has identified plans to develop a PCP, however the NRC staff is not making any regulatory findings in this SE pertaining to the PCP or for any of the transportation requirements that a PCP would address in 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," 10 CFR 71.43, "General standards for all packages," 10 CFR 71.47, "External radiation standards for all packages," 10 CFR 71.87, "Routine determinations," and 10 CFR 20.1904, "Labeling containers," as the PCP relates to classifying, processing, and disposing of radioactive waste in approved containers and packaging methods. These requirements are not applicable to a CP application, and the applicant is not seeking to demonstrate compliance with these regulations at the CP application stage. The NRC staff's evaluations of the PCP will be performed in its review of any OL application.

9.3.2 Conclusion

Based on its evaluation above, the NRC staff concludes that the information in the LMGS PSAR section 9.3 is sufficient and meets the applicable guidance and regulatory requirements identified in this chapter for the issuance of a CP in accordance with 10 CFR 50.34a(a) and PDCs 60 and 61.

9.4 Radiation Monitoring

PSAR section 9.4 describes the Radiation Monitoring System (RMS). The RMS uses radiation detectors to monitor and log radiological parameters related to the operation of the facility. The RMS includes two primary types of radiation detectors: radiation area monitors and continuous air monitors.

9.4.1 Technical Evaluation

As discussed in DANU-ISG-2022-03, the purpose of the NRC staff's review of radiation monitoring is to ensure that the CP application considers the design and programmatic elements that are required as part of an OL application that may benefit from early assessment and description in the PSAR. To demonstrate compliance with the requirements of 10 CFR Part 20, an RMS is used to measure, evaluate, and report the radiological conditions and effluent releases from a facility. PDC 63 specifies that appropriate systems shall be provided for radioactive waste systems to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels, and to initiate appropriate safety actions. PDC 64 specifies the monitoring requirements for the effluent discharge paths for radioactivity that may be released from normal operations, including anticipated operational occurrences and design basis events. For the LMGS facility, this pathway is only for the release of gaseous effluents. As discussed in section 1.3.1 of this SE, the NRC staff's review includes evaluating compliance of the LMGS design against the technically relevant portions of 10 CFR 50.34(f). With respect to the LMGS CP application and radiation monitoring, the applicant identified the technically relevant portion of 10 CFR 50.34(f)(2) as (xxvii). As specified in 10 CFR 50.34(f)(2)(xxvii), an applicant shall provide for monitoring of in-plant radiation and airborne radioactivity, as appropriate, for a broad range of routine and accident conditions.

During a CP review, the NRC staff reviews the application to check for any required safety-related or non-safety-related with special treatment radiation monitoring equipment to ensure that the requirements in 10 CFR 50.34(f)(2)(xxvii) are met, and confirms that effluent discharge pathways include monitoring of plant effluents.

The NRC staff reviewed PSAR section 9.4.1, which states that the RMS design function is to measure radiological parameters, record and display radiological data for operations and analysis, and generate alarms when radiological setpoints are exceeded to meet the applicable regulatory limits. The RMS monitors plant areas for gamma and beta radiation, airborne particulates, as well as halogen, noble gas, and tritium activity. This PSAR section also states that the RMS does not have safety-significant probabilistic risk assessment safety functions.

The NRC staff reviewed PSAR section 9.4.3, which states that the RMS provides radiation monitors that monitor plant areas for gross gamma, beta, and airborne particulates. This section also states that these area monitors are provided to support design features that meet RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants" (ML18136A762), RG 8.2, "Administrative Practices in Radiation Surveys and Monitoring" (ML110460093), and RG 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable" (ML003739549).

The NRC staff reviewed PSAR section 9.1.3, which states that plant-wide radiation monitoring will be covered by the RMS. The GASS will have its own designated radiation monitoring equipment for monitoring radioactivity in the process helium. In addition, PSAR section 9.1.3 states that the details of the HSS radiation monitoring requirement and associated design will be presented to support any OL application.

The NRC staff reviewed PSAR section 9.4.4, which states that the primary means of assessing the isotopic activities in effluent paths will be a site-specific program of sampling and onsite laboratory measurements. Detailed information on effluent monitoring and sampling programs, radiation monitors, and requirements will be provided in any future OL application.

The NRC staff reviewed PSAR table 1.4-2, which states that the guidance of RG 1.21 is not applicable to the design and construction phase of the project and will be addressed in any future OL application. The NRC staff review finds that RG 1.21, Revision 3 is appropriate guidance for describing methods for measuring, evaluating, and reporting radioactive material in liquid and gaseous effluents and solid waste; and is appropriate guidance for assessing and reporting public dose to demonstrate compliance with 10 CFR Part 20 and 40 CFR Part 190 dose limits and concentration limits, as well as showing how the 10 CFR Part 50, Appendix I dose objectives are met. The NRC staff determined that RG 1.21, Revision 3 is appropriate guidance because reference to RG 1.21 provides insights into the monitoring program that LMGS plans to implement and provides reasonable assurance that the monitoring program will follow approved NRC methods. As stated in PSAR section 9.4.1, the RMS supports the monitoring of releases to ensure releases below the regulatory limits. The NRC staff finds that the information provided in the PSAR adequately ensures that monitoring is considered in the CP application to track radioactive materials, with additional details to be provided during the OL stage which will need to show how the guidance is met.

In support of PDC 63, LME provided high-level descriptions in PSAR chapters 9 and 10 stating that area radiation monitoring will be available throughout the plant to measure radiological conditions of the facility. To address PDC 64 and 10 CFR 50.34(f)(2)(xxvii), LME provided discussions in PSAR sections 9.1.3 and 9.4.4 to describe preliminary information about the planned effluent monitoring and helium monitoring. The information provided in the PSAR supports the CP application and further information will be provided and reviewed by the NRC staff during its review of any OL application.

The NRC staff determined that the preliminary information describing planned monitoring to support the control of radioactive materials is sufficient for the CP application because LME has provided high-level descriptions of monitoring locations and has stated that LME will describe how the guidance of RG 1.21 will be met at the OL stage. In addition, 10 CFR 50.34(f)(2)(xxvii) and PDCs 63 and 64 are met because preliminary information was provided to address plans for radiation monitoring, including placement in various areas of the facility to monitor radiological conditions and availability to monitor gaseous effluent releases. The NRC staff will review the final radiation monitoring information at the OL stage to ensure compliance with 10 CFR 50.34(f)(2)(xxvii) and PDCs 63 and 64.

9.4.2 Conclusion

Based on its evaluation above, the NRC staff concludes that the information in the LMGS PSAR section 9.4 is sufficient and meets the applicable guidance and regulatory requirements identified in this chapter for the issuance of a CP in accordance with 10 CFR 50.34(f)(2)(xxvii), and PDCs 63 and 64.

9.5 Contamination Control

PSAR section 9.5 describes design features and procedures that minimize, to the extent practicable, contamination of the facility and the environment and facilitate eventual decommissioning, as well as minimize, to the extent practicable, the generation of radioactive waste consistent with the requirements in 10 CFR 20.1406, “Minimization of contamination.”

9.5.1 Technical Evaluation

As discussed in DANU-ISG-2022-03, the purpose of the NRC staff’s review of contamination control is to ensure that the CP application describes sufficient design and programmatic information to address 10 CFR 20.1406, which requires that the design and operation minimize contamination of the facility and the environment. For the review of contamination control, the applicant must meet the regulations specified by 10 CFR 20.1406 by describing how the applicant intends to minimize, to the extent practicable, the contamination of the facility and the environment, and the generation of radioactive waste.

The intent of 10 CFR 20.1406 is to emphasize the importance of designing and operating facilities in a way that would minimize the amount of radioactive contamination generated at the site during the operating lifetime and would minimize the generation of radioactive waste during decommissioning. As provided in 10 CFR 20.1002, “Scope,” the regulations in 10 CFR Part 20 apply to “persons licensed by the Commission to receive, possess, use, transfer, or dispose of byproduct, source, or special nuclear material [SNM] or to operate a production or utilization facility….” LME has applied for a CP, which does not provide a license to operate the facility. In the CP application, LME also has not applied for licenses to receive, possess, use, transfer, or dispose of any byproduct, source, or SNM at the facility. Therefore, the NRC staff did not evaluate whether the requirements in 10 CFR Part 20 would be met in support of the CP. Instead, the NRC staff assessed whether LME had identified the relevant requirements for an operating facility and provided descriptions of how the preliminary facility design will, to the extent practicable, minimize contamination of the facility and the environment, as required for operating facilities under 10 CFR 20.1406. This is consistent with 10 CFR 50.40(a), which provides that in determining whether a CP may be issued, the Commission will be guided by consideration of reasonable assurance that LME will comply with the NRC’s regulations,

including the regulations in 10 CFR Part 20, and that the health and safety of the public will not be endangered. For the CP application review, the NRC staff ensured that the applicant has met 10 CFR 50.40 and has provided enough information to provide reasonable assurance that the applicant will comply with the regulations in 10 CFR 20.1406 at the OL stage.

The NRC staff reviewed PSAR section 9.5, which states that the LMGS plant design, programs, and procedures have the goals of minimizing, to the extent practicable, contamination of the plant facility and the environment. Plant SSCs with the potential to contain/handle radiological materials will include design considerations to minimize contamination. LME states that the design features, programmatic elements, and procedures will consider the guidance of RG 4.21. A description of the design features and procedures for the control of radioactive contamination for the plant, including consideration of RG 4.21, will be provided at the OL stage.

The NRC staff has determined that the LMGS PSAR section 9.5 on contamination control is acceptable because LME will consider the guidance in RG 4.21 and because LME will provide further details on the design features as well as procedures for the control of radioactive contamination to support an OL application. LME's consideration of RG 4.21 provides the NRC staff with reasonable assurance that the requirements of 10 CFR 20.1406 will be met. The NRC staff will review the details of the design features considered at the OL stage to ensure compliance with 10 CFR 20.1406.

9.5.2 Conclusion

Based on its evaluation above, the NRC staff concludes that the information in the LMGS PSAR section 9.5 is sufficient and meets the applicable guidance and regulatory requirements identified in this section for the issuance of a CP in accordance with 10 CFR 50.40.

9.6 Conclusion

Based on its findings above, the NRC staff concludes that the information in PSAR chapter 9 is sufficient and meets the applicable guidance and regulatory requirements identified in this chapter for the issuance of a CP in accordance with 10 CFR 50.34a(a), 10 CFR 50.34a(b)(1), 10 CFR 50.34a(b)(2), 10 CFR 50.34(f)(2)(xxvii), and 10 CFR 50.40.

10 CONTROL OF OCCUPATIONAL DOSE

This chapter of the safety evaluation (SE) summarizes the U.S. Nuclear Regulatory Commission (NRC or Commission) staff's review and evaluation of the Long Mott Generating Station (LMGS) preliminary safety analysis report (PSAR) chapter 10, "Control of Occupational Dose," which contains preliminary descriptions of the radiation protection program (RPP), the as low as is reasonably achievable (ALARA) program, design considerations and design features to maintain occupational radiation exposures ALARA, and the preliminary dose assessment for expected occupancy performed for the construction permit (CP) application.

The applicable regulatory requirements for the evaluation of the control of occupational dose are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a)(6); and
- 10 CFR 50.40, "Common standards."

The applicable guidance for the evaluation of the control of occupational dose is as follows:

- Division of Advanced Reactors and Non-Power Production and Utilization Facilities (DANU) Interim Staff Guidance (ISG) DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications - Roadmap," section 12 (ML23277A139);
- DANU-ISG-2022-04, "Advanced Reactor Content of Application Project, Chapter 10, 'Control of Occupational Dose'" (ML23277A142);
- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light Water Reactor] Edition," section 12.5 (ML13155A232).

The following principal design criteria (PDC), as defined in PSAR section 5.3, apply to the evaluation of the control of occupational dose:

- PDC 19, "Control Room";
- PDC 61, "Fuel Storage and Handling and Radioactivity Control."

Paragraph 50.34(a)(6) of 10 CFR requires that a CP application contain a preliminary plan for the applicant's organization, training of personnel, and conduct of operations. Consistent with this requirement, DANU-ISG-2022-04 states that a CP application should include: (1) a description of its plans to develop, at the operating license (OL) application stage, comprehensive worker protection programs, organizational structure, training, and monitoring to ensure the requirements are met in 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations," and 10 CFR Part 20, "Standards for Protection Against Radiation"; (2) a description of the design-specific PDC necessary to control occupational exposures; and (3) a preliminary description of design provisions necessary to control occupational exposure, including ensuring that occupational doses are ALARA, such as

shielding, ventilation, area radiation and airborne radioactivity monitoring instrumentation, and dose assessment for expected occupancy.

The NRC staff's evaluation of the requirements in 10 CFR Part 20, will be conducted during the OL application review consistent with 10 CFR 50.34(b)(3), which provides the OL application requirement to provide the kinds and quantities of radioactive materials expected to be produced during operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in 10 CFR Part 20.

Section 20.1101, "Radiation protection programs," of 10 CFR, states that a "licensee" shall develop, document, and implement an RPP commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the provisions of 10 CFR Part 20 and shall use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses that are ALARA. As provided in 10 CFR 20.1002, "Scope," the regulations in 10 CFR Part 20 apply to "persons licensed by the Commission to receive, possess, use, transfer, or dispose of byproduct, source, or special nuclear material or to operate a production or utilization facility…." Since Long Mott Energy, LLC (LME or the applicant) is not a licensee within the scope of 10 CFR Part 20, the NRC staff did not evaluate whether the requirements in 10 CFR Part 20 would be met in support of its review of the CP application. Instead, consistent with DANU-ISG-2022-04, the NRC staff assessed whether LME had identified the relevant requirements for an operating facility and provided descriptions of how the preliminary facility design will, to the extent practicable, meet the 10 CFR Part 20 requirements as discussed in the technical evaluation. This is consistent with 10 CFR 50.40(a), which provides that in determining whether a CP will be issued, the Commission will be guided by, among others, consideration of reasonable assurance that LME will comply with the NRC's regulations, including the regulations in 10 CFR Part 20, and that the health and safety of the public will not be endangered. For the CP application review, the NRC staff ensures that LME has met 10 CFR 50.40 and has provided enough information to provide reasonable assurance that it will comply with the regulations in 10 CFR Part 20 at the OL stage.

10.1 Radiation Protection Program

As discussed in DANU-ISG-2022-04, the purpose of the NRC staff's review of the RPP is to ensure that the CP application includes, among other things, in accordance with 10 CFR 50.34(a)(6), a description of LME's plans to develop, at the OL stage, comprehensive worker protection programs, organizational structure, training, and monitoring to ensure that the relevant requirements are met in 10 CFR Parts 19 and 20. The NRC staff's evaluation of the overall organization as required to meet 10 CFR 50.34(a)(6) is addressed in section 11.1 of this SE.

10.1.1 Technical Evaluation

The NRC staff reviewed the LMGS PSAR to determine whether, as required by 10 CFR 50.34(a)(6), the applicant for the CP adequately described its plans to develop an RPP. The NRC staff reviewed PSAR chapter 10, which states that the RPP will be developed and implemented, as required by 10 CFR 20.1101, prior to receiving radioactive materials on-site and that the RPP will ensure that 10 CFR 19.12, "Instruction to workers," and 10 CFR Part 20 requirements are met. PSAR section 10.1 references Nuclear Energy Institute (NEI) 07-03A, "Generic FSAR [Final Safety Analysis Report] Template Guidance for Radiation Protection Program Description," (ML091490684) and NEI 07-08A, "Generic FSAR Template Guidance for

Ensuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable (ALARA)," (ML093220178) and states that the RPP will be consistent with the guidance in NUREG-1736, "Consolidated Guidance: 10 CFR Part 20 - Standards for Protection Against Radiation," (ML013330106) and states full conformance or partial conformance with Regulatory Guide (RG) 1.8, "Preoperational Testing of Instrument Air Systems" (ML19101A395), RG 8.2, "Administrative Practices in Radiation Surveys and Monitoring" (ML110460093), RG 8.4, "Personnel Monitoring Device—Direct-Reading Pocket Dosimeters" (ML101900087), RG 8.7, "Instructions for Recording and Reporting Occupational Radiation Exposure Data" (ML17221A245), RG 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable" (ML003739549), RG 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable" (ML16105A136), RG 8.15, "Acceptable Programs for Respiratory Protection" (ML003739528), RG 8.27, "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants" (ML003739628), RG 8.34, "Monitoring Criteria and Methods To Calculate Occupational Radiation Doses" (ML22132A083), and RG 8.38, "Control of Access to High and Very High Radiation Areas of Nuclear Plants" (ML061350096), which are also listed in PSAR table 1.4-2.

In addition, PSAR section 10.1 describes that the purpose of the RPP is to maintain occupational dose below regulatory limits and keep exposure ALARA; have a trained and qualified organization; and have adequate facilities, equipment, and procedures to effectively implement the program for various functional areas. This PSAR section also describes other areas that the RPP will encompass. In PSAR section 10.1.1, LME describes that the RPP meets the requirements outlined in 10 CFR Part 20, Subpart F and RG 8.2, Rev. 1, and states that additional details on radiation monitor locations will be provided at the OL stage. In PSAR section 10.1.2, LME describes the controls for access to high and very high radiation areas and states that they will conform with RG 8.38, Revision 1. PSAR table 10.5-2 describes the zoning designations that will be applied to the LMGS.

DANU-ISG-2022-01 references NUREG-0800, section 12.5, which provides additional guidance for an RPP and includes a list of RGs that an RPP should consider. Although NUREG-0800, section 12.5 is specific to light-water reactors (LWR) applications, it may provide useful insights for the NRC staff's review of non-LWR applications. Consistent with the guidance in NUREG-0800, section 12.5, the NRC staff's review determined that the high-level description of the LMGS RPP and LME's commitments to follow applicable guidance are appropriate at the CP stage. This approach is acceptable because LME has identified those elements of an RPP that are considered at the CP stage and has committed to following additional RGs, which will be evaluated at the OL stage. The NRC staff's review of PSAR table 1.4-2 determined that LME has either stated that LMGS is in full conformance with or will address conformance with applicable RGs that are a part of an RPP in any OL application.

The NRC staff's review of PSAR section 10.1 finds that LME provided a high-level overview of the RPP and that the references to various RGs are appropriate for the CP stage because they demonstrate areas that an RPP should include for future consideration in an OL application, and the information provided in the CP application is consistent with the review guidance specified by DANU-ISG-2022-01 and the RGs listed in NUREG-0800, section 12.5.

Radiation protection training is addressed by the contents of PSAR table 1.4-2, which provides that conformance with RG 8.27, Revision 0, will be addressed at the OL stage. RG 8.27, Revision 0 describes a method to develop a radiation protection training program consistent

with ALARA objectives that is acceptable to the NRC staff for meeting 10 CFR 19.12. This is acceptable because LME is not required to address the operational training program for radiation protection during the CP stage, and the NRC staff will review the details of radiation protection training during the review of any OL application.

The radiation protection organizational structure is addressed by LME's commitments to RG 8.8, Revision 3 and RG 8.10, Revision 2. RG 8.8, Revision 3 and RG 8.10, Revision 2 describe methods of meeting 10 CFR Part 20 requirements, which include aspects of an organization that address ALARA, that are acceptable to the NRC staff. In addition, PSAR table 1.4-2 and PSAR section 10.1 describe that LME will address compliance with RG 1.8, Revision 4 during the OL stage for the qualification of radiation protection personnel. RG 1.8, Revision 4 describes a method acceptable to the NRC staff for complying with regulations associated with the organizational structure under 10 CFR 50.34(b)(6)(i), which would address the requirements in 10 CFR 50.34(a)(6) for a CP application. Based on this information, the NRC staff finds that LME has included sufficient information about its radiation protection organizational structure for the CP stage, which includes providing preliminary commitments to RG 8.8, Revision 3 and RG 8.10, Revision 2, and that it is appropriate to leave the remaining details to the OL stage. The NRC staff's review of the overall organization, including the radiation protection organization, is summarized in SE section 11.1.

10.1.2 Conclusion

The NRC staff's review of the RPP determined that the high-level description of the RPP and the commitments to address compliance with the RGs discussed above are sufficient because LME referenced RGs for the RPP consistent with DANU-ISG-2022-01 and NUREG-0800, section 12.5. The NRC staff's review of the RPP can be reasonably deferred to the OL stage because details regarding specific aspects of the RPP and procedures are not required at the CP stage. The NRC staff will review LMGS RPP information at the OL stage to ensure compliance with 10 CFR 19.12, 10 CFR Part 20, and 10 CFR 50.34(b)(6)(i).

The NRC staff determined that referencing RGs 1.8 and 8.27 provides the NRC staff with approved methods that LME will use to qualify and train personnel for nuclear power plants and radiation protection training for personnel at nuclear power plants and is, therefore, acceptable. Accordingly, the NRC staff determined that the applicant has provided sufficient information for a CP application to describe the preliminary plan for the applicant's RPP organization, training of personnel, and conduct of operations as required by 10 CFR 50.34(a)(6).

10.2 ALARA Program

As discussed in DANU-ISG-2022-04, the purpose of the NRC staff's review of the ALARA program is to ensure that the CP application includes, among other things, in accordance with 10 CFR 50.34(a)(6), a description of LME's plans to develop, at the OL stage, comprehensive worker protection programs, organizational structure, training, and monitoring to ensure that the relevant requirements are met in 10 CFR Part 20.

10.2.1 Technical Evaluation

The NRC staff reviewed PSAR section 10.2, which states that as part of the LMGS RPP and in accordance with 10 CFR Part 20, maintaining occupational doses below regulatory limits and ALARA is a key focus of the LMGS plant structures, systems, and components and

programmatic design and implementation activities. PSAR section 10.1 references RG 8.10, which provides guidance for establishing an ALARA program that is acceptable to the NRC staff, and PSAR table 1.4-2 states that LMGS is in partial conformance with RG 8.10, Revision 2 because the guidance is only applicable to LWRs is therefore not applicable to LMGS. In PSAR section 10.2, LME states that additional details of the ALARA program will be provided in any OL application. The NRC staff finds that the applicant's use of RG 8.10, Revision 2 is appropriate because it is the applicable guidance and it will be followed with the exception of its guidance that is only applicable to LWRs. The NRC staff also finds that that LWR-specific guidance relates to application-specific guidance that would be otherwise met through the use of DANU-ISG-2022-04.

10.2.2 Conclusion

The NRC staff determined that the description provided in the CP application of LME's plan to develop an ALARA program at the OL stage is acceptable because LME states that the ALARA program will be in partial conformance with RG 8.10, as described above. The NRC staff will review the LMGS ALARA program as part of the review of any OL application to ensure compliance with relevant regulations.

10.3 Design Considerations

As discussed in DANU-ISG-2022-04, one purpose of the NRC staff's review is to ensure that a CP application provides a description of the design-specific PDCs necessary to control occupational exposures and a preliminary description of design provisions necessary to control occupational exposure, including ensuring that occupational doses are ALARA. As discussed in SE section 1.4.1, the NRC staff's review includes evaluating compliance of the LMGS design with the technically relevant portions of 10 CFR 50.34(f), which are identified in PSAR table 1.4-1. With respect to the LMGS CP application and the control of occupational dose, the applicant identified the technically relevant portions of 10 CFR 50.34(f)(2) as (vii), (xxvii), and (xxviii). As specified in 10 CFR 50.34(f)(2)(vii), an applicant shall perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. As specified in 10 CFR 50.34(f)(2)(xxvii), an applicant shall provide for monitoring of in-plant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions. As specified in 10 CFR 50.34(f)(2)(xxviii), an applicant shall evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source term release and make necessary design provisions to preclude such problems. The regulation at 10 CFR 50.34(f)(2) states that to satisfy these requirements, CP applications must provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the OL stage.

10.3.1 Technical Evaluation

The NRC staff reviewed PSAR section 10.3.1, which cites the relevant radiation protection aspects of PDCs 19 and 61, which are the design-specific PDCs necessary to control occupational exposures. The PDCs for LMGS are provided in the PSAR and details on the PDC development process and basis are in the topical report, "Xe-100 Licensing Topical Report Principal Design Criteria," Revision 3 (ML24319A155). The NRC staff's review of PDCs is summarized in SE section 5.3. PSAR section 10.3.2 and table 10.3-1 provide preliminary

information on the major radiation sources expected at the facility. PSAR table 1.4-1 provides that 10 CFR 50.34(f)(2)(vii), (xxvii), and (xxviii) are addressed in full compliance.

The NRC staff reviewed PSAR section 10.3, which describes that a key objective of the LMGS design is to reduce radiation fields in operating areas and that ALARA principles are applied throughout design processes via methodologies and design decisions. In PSAR section 10.3, LME provides that the radiation protection design approach includes overarching design approaches, system level considerations, plant features, and mitigation efforts to minimize dose and maintain ALARA in accordance with, but not limited to, 10 CFR 20.1101. LME also provides that detailed design information will be available during the review of any OL application. PSAR section 10.3.2 provides a list of preliminary radiation sources in PSAR table 10.3-1. PSAR section 10.3.2 also states that the sources in this table range from major radiation sources, such as spent fuel, to more limited sources, such as filters and dust. Additional details of the radiation sources for the LMGS needed to demonstrate compliance with 10 CFR Part 20 will be provided to support any OL application.

The NRC staff reviewed the information in PSAR section 10.3.2 and table 10.3-1 to understand the locations of potential radiation sources. To support its understanding of these radiation sources, the NRC staff conducted an audit of the facility's preliminary estimates for dose (ML26293A340). During the audit, LME provided the NRC staff with insights into its high-level planning for anticipated radiation zoning levels consistent with the information provided in PSAR table 10.3-1.

For the LMGS CP application, the NRC staff verified that LME has committed to the methods contained in RG 8.8, Revision 3 and other RGs applicable to the RPP. RG 8.8 describes information relevant to attaining goals and objectives for planning, designing, constructing, operating, and decommissioning an LWR to meet the ALARA criteria, but also contains information applicable to non-LWR designs. Commitments to this RG along with the other RGs discussed above with respect to the RPP provide reasonable assurance that ALARA design considerations will be included as the LMGS design is being finalized. The NRC staff's assessment of the preliminary information provided to meet 10 CFR 50.34(f)(2)(xxvii) is in SE section 9.4. Additional information to support the NRC staff's review of ALARA design considerations and dose assessments would be evaluated as part of the review of any OL application. For example, an OL application review would include evaluating the ALARA design considerations against the requirements of 10 CFR Part 20, 10 CFR 50.34(b)(3), 10 CFR 50.34(f)(2)(vii), 10 CFR 50.34(f)(2)(xxvii), 10 CFR 50.34(f)(2)(xxviii), PDC 19, and PDC 61.

10.3.2 Conclusion

Given PDCs 19 and 61 and the commitments to follow the RGs described above for the RPP and the ALARA program and to fully comply with 10 CFR 50.34(f)(2)(vii), (xxvii), and (xxviii), the NRC staff has reasonable assurance that LME will be able to incorporate design considerations such as shielding into the final design to ensure that dose estimates and radiation zones are maintained at appropriate levels for the LMGS facility. The NRC staff would review the design considerations, radiation sources, and dose assessments at the OL stage to ensure compliance with 10 CFR Part 20, 10 CFR 50.34(b)(3), 10 CFR 50.34(f)(2)(vii), 10 CFR 50.34(f)(2)(xxvii), 10 CFR 50.34(f)(2)(xxviii), PDC 19, and PDC 61.

10.4 Design Features

As discussed in DANU-ISG-2022-04, one purpose of the NRC staff's review is to ensure that a CP application provides a preliminary description of design provisions necessary to control occupational exposure, including ensuring that occupational doses are ALARA.

10.4.1 Technical Evaluation

The NRC staff reviewed PSAR section 10.4, which states that plant design features, such as ALARA considerations during shielding design and remote operations where feasible, are considerations made during the design phase and maintain occupational radiation exposure ALARA. PSAR section 10.4.1 states that the design of the plant considers adequate shielding design to support occupational radiation exposure ALARA and within the limits of 10 CFR Part 20. In this PSAR section, the applicant also highlights the regulatory requirements, regulatory guidance, and applicable codes and standards considered during the shielding design process for the plant. PSAR table 1.4-2 further provides that LMGS will be in full conformance with RG 1.69, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants," Revision 1 and RG 8.8, Revision 3. PSAR section 10.4.2 states that the design considers remote operation activities where available, such as remote monitoring and controlling of various plant parameters and autonomous or remotely operated special tooling for inspection, service, and maintenance to minimize dose to plant staff to achieve ALARA. This section also provides some examples of remotely operated tooling operations and manipulations, which are installing and removing components, turning valves, manipulating switches, racking breakers, connecting and disconnecting connectors, and using hand-tools.

The NRC staff reviewed the information cited as applicable guidance for shielding considerations and finds that the references to RG 1.69 for the design and construction of shielding and to RG 8.8 for ALARA design considerations, are appropriate for discussing design features that will be considered at the LMGS because RGs 1.69, Revision 1 and 8.8, Revision 3 describe acceptable methods for meeting shielding design requirements and ALARA shielding design features.

10.4.2 Conclusion

The NRC staff determined that the radiation protection design features and commitments to conform with RG 1.69, Revision 1 and RG 8.8, Revision 3 are appropriate because LME has committed to use NRC-approved methods for determining applicable design features to consider at the LMGS and has provided the NRC staff with a preliminary description of the design provisions necessary to control occupational exposure that would address the guidance for CP applications in DANU-ISG-2022-04.

10.5 Preliminary Dose Assessment for Expected Occupancy

As discussed in DANU-ISG-2022-04, one purpose of the NRC staff's review is to ensure that a CP application provides a preliminary description of design provisions necessary to control occupational exposure, including ensuring that occupational doses are ALARA.

10.5.1 Technical Evaluation

The NRC staff did not conduct a detailed review of PSAR section 10.5, which provides preliminary dose estimates for occupational workers, because this information is not required for a CP application. This section states that the estimated radiation exposure to plant staff was not based on expected source terms and does not represent the actual dose rates anticipated at the LMGS. PSAR section 10.5 also states that detailed task-based dose assessments for plant staff, considering plant radiation zoning where plant staff would be expected to perform normal duties and staff occupancy information, and information similar to the tables presented in RG 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants -- Design Stage Man-Rem Estimates," would be provided to support any OL application. Thus, initial radiation zones were developed to provide a basis for the preliminary design of the plant. The NRC staff reviewed the design considerations and design features as discussed in SE sections 10.3.1 and 10.4.1, respectively, which provide information on the preliminary design considerations and features that would inform the occupational dose assessment that would be performed at the OL stage.

10.5.2 Conclusion

With LME's commitment to follow RG 8.8 to manage dose, as discussed in SE sections 10.1 through 10.4, and the identification of major radiation sources, the NRC staff has determined that LME has identified the major radiation sources and will consider design considerations and design features as appropriate to ensure that occupational exposures are within regulatory limits and maintained ALARA. The commitments to follow the RGs discussed in this SE chapter provide the preliminary description of design provisions necessary to control occupational exposure, including ensuring that occupational doses are maintained ALARA.

The NRC staff would review the occupational dose assessment at the OL stage when the details required to assess dose, such as radiation sources, would be provided, as required by 10 CFR 50.34(b)(3). The NRC staff would also review the LMGS occupational dose assessment at the OL stage to ensure compliance with 10 CFR Part 20.

10.6 Conclusion

Based on its findings above, the NRC staff concludes that the information in PSAR chapter 10 is sufficient and meets the applicable guidance and regulatory requirements identified in this chapter for the issuance of a CP in accordance with 10 CFR 50.34(a)(6) and 10 CFR 50.40.

11 CONDUCT OF OPERATIONS

This chapter of the preliminary safety analysis report (PSAR) describes the preliminary plan for the applicant's organization, training of personnel, conduct of operations, and coping with emergencies that are required by 10 CFR 50.34(a)(6) and (10) to be included in the construction permit (CP) application. This chapter also provides information related to the technical qualifications of the applicant to engage in the activities discussed in the application pursuant to 10 CFR 50.34(a)(9). Also included in this chapter is a description of the construction fitness-for-duty (FFD) program required by 10 CFR 26.401(b).

This chapter is structured as follows:

- Section 11.1 - Organization
- Section 11.2 - Concept of Operations
- Section 11.3 - Training
- Section 11.4 - Emergency Planning
- Section 11.5 - Procedures
- Section 11.6 - Security and FFD

The information in this chapter is informed by the Advanced Reactor Content of Application Project (ARCAP) interim staff guidance - for example, DANU-ISG-2022-01, "Review of Risk-Informed, Technology-inclusive Advanced Reactor Applications – Roadmap," and DANU-ISG-2022-05, "Advanced Reactor Content of Application Project Chapter 11, 'Organization and Human-System Considerations,'" - that the U.S. Nuclear Regulatory Commission (NRC) has issued to support the development and review of a CP application PSAR for an advanced non-light-water reactor (non-LWR). The human factors engineering program considerations recommended in the NRC ARCAP guidance for PSAR chapter 11 are described in PSAR section 8.7 and evaluated in SE section 8.7.

11.1 Organization

PSAR section 11.1 describes the applicant's organizational structure and key management positions in the organizations that are responsible for facility design, design review, design approval, construction management, testing, and plant operation.

The applicable regulatory requirements for the evaluation of the design, construction, and operating organizations are as follows:

- 10 CFR 50.34(a), which describes the minimum information to be included in a PSAR and requires, in part, the following:
    - 50.34(a)(6) - A preliminary plan for the applicant's organization, training of personnel, and conduct of operations.
    - 50.34(a)(9) - The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter.

- 10 CFR 50.34(f)(2)(i), which requires, in part, that CP applications provide sufficient information to demonstrate that the simulator capability correctly models small-break LOCAs.

- 10 CFR 26.3(c), which requires, in part, that CP holders shall comply with the requirements of 10 CFR Part 26, Subpart I, “Managing Fatigue,” by no later than the receipt of special nuclear material in the form of fuel assemblies.

The applicable guidance for the evaluation of this PSAR section is as follows:

- DANU-ISG-2022-01, “Review of Risk-Informed, Technology-inclusive Advanced Reactor Applications – Roadmap” Interim Staff Guidance,” March 2024, (ML23277A139);

- DANU-ISG-2022-05, “Advanced Reactor Content of Application Project Chapter 11, ‘Organization and Human-System Considerations’ Interim Staff Guidance,” March 2024, (ML23277A143);

- NUREG-0800 Chapter 13.1.1, "Management and Technical Support Organization,” Revision 6, August 2016, (ML15005A449);

- NUREG-0800 Chapter 13.1.2 - 13.1.3, "Operating Organization,” Revision 7, August 2016, (ML15007A296);

- NUREG-0800 Chapter 13.5.1.1, “Administrative Procedures – General,” Revision 2, August 2016, (ML15006A205);

- NUREG-0800 Chapter 13.5.2.1, “Operating and Emergency Operating Procedures,” Revision 2, March 2007, (ML070100635);

- Regulatory Guide (RG) 1.8, “Qualification and Training of Personnel for Nuclear Power Plants,” Revision 4, June 2019, (ML19101A395);

- RG 1.28, “Quality Assurance Program Criteria (Design and Construction),” Revision 5, (ML17207A293);

- RG 1.33, “Quality Assurance Program Requirements (Operation),” Revision 3, June 2013, (ML13109A458);

- RG 1.114, “Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit,” Revision 3, October 2008, (ML082380236);

- American National Standards Institute/American Nuclear Society (ANSI/ANS)-3.1-2014, “Selection, Qualification, and Training of Personnel for Nuclear Power Plants”;

- ANSI/ANS-3.2-2012, “Managerial, Administrative, and Quality Assurance Controls for the Operational Phase of Nuclear Power Plants.”

11.1.1 Technical Evaluation

The NRC staff evaluated PSAR section 11.1, 11.2, 11.3, and 11.5 using the guidance and criteria of DANU-ISG-2022-05.

11.1.1.1 Assignment of Plant Operating Responsibilities

DANU-ISG-2022-05 criterion 11.2(a) states that the applicant should describe the assignment of plant operating responsibilities, the reporting chain up through the chief executive officer of the applicant, the proposed size of the regular plant staff, the functions and responsibilities of each major plant staff group, the proposed shift crew complement for single-module/-unit or multiple-module/-unit operation, and the qualification requirements for key members of the plant staff.

The NRC staff reviewed PSAR section 11.1.1, which describes the functions and responsibilities of each major plant staff group that will be established in support of construction and preoperational testing activities. The reporting chain up through the chief executive officer is depicted in PSAR figure 11.1-1, including, in part, engineering design, procurement and supply chain, technology development, and program director reporting to the senior vice president of commercial operations. The senior vice president of commercial operations reports to the president, along with the quality assurance manager and environmental health and safety.

With specific regard to operations staffing, PSAR section 11.2.3 states that the requirements of 10 CFR 50.54(m) will be addressed at the operating license (OL) application stage. The NRC staff notes that the specifics of the shift crew complement that will be utilized during plant operations is not required until an OL application. The NRC staff further notes that, although the proposed size of the regular plant staff is not described in numerical terms, the PSAR references the NRC-approved “Xe-100 Licensing Topical Report Control Room Staffing Analysis Methodology” (ML23258A243), which would be used to develop the staffing plan described in the application for an OL. The NRC staff’s review of ML23258A243 notes that future proposed staffing levels will be prepared using the Xe-100 human factors engineering (HFE) Program, which serves as the basis for control room staffing in accordance with the guidance in NUREG-1791, “Guidance for Assessing Exemption Request from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m).” PSAR section 11.2.3 further confirms the applicant’s intent to develop a staffing plan with an analysis sufficient to justify a request for an exemption to 10 CFR 50.54(m) which would be submitted as part of the OL application. Accordingly, the NRC staff finds that the descriptions of the organizational structure, reporting relationships, and functions across the organization provide an appropriate degree of detail for the CP application review stage, as discussed in DANU-ISG-2022-05, and that the review of further details regarding operating organization composition is not required until the review of an OL application. The NRC staff therefore concludes that the applicant has adequately described plant operating responsibilities, reporting relationships, organizational structure, functional areas, and qualifications to the degree appropriate for a PSAR, consistent with criterion 11.2(a) presented in DANU-ISG-2022-05, and further analysis is not required until the review of an OL application.

11.1.1.2 Technical Qualification of Applicant

DANU-ISG-2022-05, criterion 11.2(b) states, in part, that the applicant should be technically qualified. The NRC staff reviewed PSAR section 11.1, which states that Long Mott Energy, LLC (LME), along with X Energy, LLC (X-energy), is responsible for the design and construction of the Long Mott Generating Station (LMGS). LME is a wholly owned subsidiary of GWN Holding, LLC, which is a wholly owned subsidiary of The Dow Chemical Company (TDCC). TDCC is described as a large multi-national petrochemical company that safely operates more than 50 gas and steam turbines at 30 sites worldwide. TDCC is also described as having expertise and experience in design and construction management of large-scale projects. TDCC and LME have an inter-company agreement to procure technical and financial support, including personnel, to ensure that the requisite leadership, engineering, procurement and administrative assistance is available for completion of the design and construction of LMGS. The PSAR further states that TDCC partnered with NextEra Energy Resources, LLC (NEER), a subsidiary of NextEra Energy, the largest electric utility in the United States, including the operation of seven light water reactors at four sites, to provide technical support to LME in areas including, but not limited to, nuclear procurement, licensing activities, personnel training, and operating discipline development.

Accordingly, because the PSAR discusses the technical experience available through X-energy, TDCC, and NEER, the NRC staff confirmed the PSAR is consistent with DANU-ISG-2022-05, criterion 11.2(b). Therefore, the NRC staff concludes that the applicant is technically qualified to undertake construction of LMGS, consistent with criterion 11.2(b) presented in DANU-ISG-2022-05.

11.1.1.3 Key Positions for Ensuring the Safe Operation of the Plant

DANU-ISG-2022-05, criterion 11.2(c) states that the applicant should describe the key positions for ensuring the safe operation of the plant within the operating organization, consistent with the quality assurance program (QAP) and ANSI/ANS 3.2, as well as the capability for on-shift personnel to provide initial facility response in the event of an emergency.

The NRC staff reviewed PSAR section 11.1.2.2, which discusses the responsibilities of the executive vice president and chief nuclear officer, who has overall responsibility for plant nuclear safety and implementation of the QAP.

The NRC staff reviewed PSAR section 11.1.2.3, which states that, during the operational phase, the plant general manager “assures the safe, reliable, and efficient operation of the plant within the constraints of applicable regulatory requirements, the operating license, and the QAP.”

The NRC staff reviewed PSAR figure 11.1-3, which provides an overview of the proposed operating organization, including an emergency response organization (ERO) under the operations coordinator. The NRC staff also reviewed PSAR section 11.4.3, which states that the composition, structure, and staffing of the ERO will be developed based on an analysis of the emergency response functions needed to respond to emergencies and will be submitted as part of an OL application.

The NRC staff reviewed PSAR section 12.1.1.4, which discusses the LMGS organizational plan to include construction, commissioning, and operations teams for project execution. Accordingly, because the PSAR discusses the key positions for ensuring the safe operation of the plant within the operating organization as well as the capability for on-shift personnel to provide initial facility response in the event of an emergency, the NRC staff confirmed the PSAR is consistent with the guidance in DANU-ISG-2022-05, criterion 11.2(c), and that the descriptions provided by the applicant are sufficient for the CP stage to ensure that an organizational structure will be in effect to meet the framework of the criterion. However, the NRC staff notes that the PSAR does not describe whether the operating organization will comply either with the organizational provisions of ANSI/ANS 3.2 or with an alternative approach. Given that this criterion specifically addresses the facility organization during the operating phase, the NRC staff determined that the complete evaluation of this criterion, including any related commitment to ANSI/ANS 3.2, is not required until the review of an OL application.

11.1.1.4 Initial Test Program and Technical Support

DANU-ISG-2022-05, Criterion 11.2(d) states, in part, that the applicant should adequately describe the groups and key positions responsible for implementing the initial test program (consistent with the QAP and ANSI/ANS 3.2) and for providing technical support for the operation of the facility.

The NRC staff reviewed PSAR section 12.1.1.4, which states, in part, that the commissioning schedule, in support of the initial test program, will include control room operators, plant field technicians, and other technical, engineering, and operational support personnel. Operational support personnel will include chemistry and radiation protection technicians. Additionally, the PSAR further states that vendor technical field advisors and third-party contracted resources will be available for providing support for their respective systems, structures, and components (SSCs) or discipline area.

Accordingly, because the PSAR adequately describes the groups and key positions responsible for implementing the initial test program and for providing technical support for the operation of the facility, the NRC staff confirmed the guidance in DANU-ISG-2022-05 criterion 11.2(d) is met sufficient for the CP stage to ensure an organizational structure will be in effect to meet the framework of the criterion. The NRC staff notes that ANSI/ANS 3.2 addresses the operational phase of plant life, which is entered prior to initial fuel loading. As such, the NRC staff determined that evaluation of the consistency of the test organization with either ANSI/ANS 3.2 or an alternative approach is not required until the future review of an OL application. The NRC staff therefore concludes that the applicant has described the organization responsible for implementation of the initial test program, the related QAP considerations, and the organizational provisions for the technical support of facility operations, consistent with criterion 11.2(d) presented in DANU-ISG-2022-05 as applicable for a CP, and further analysis is not required until the review of an OL application.

11.1.1.5 Plans for the Construction, Testing, and Operating Organizations

DANU-ISG-2022-05 criterion 11.2(e) states that the applicant should describe plans for the construction, testing, and operating organizations that are consistent with the relevant portions of NUREG-0800, section 13.1.1, “Management and Technical Support Organization” and

sections 13.1.2 - 13.1.3, “Operating Organization.” The NRC staff considered the guidance within these documents against the information otherwise addressed by the criteria of DANU-ISG-2022-05, as well as within the context of a CP application and identified specific criteria of NUREG-0800, section 13.1.1 that are relevant. These criteria are evaluated within sections 11.1.1.5.1 through 11.1.1.5.6 of this SE.

11.1.1.5.1 Integrated Management of Activities

NUREG-0800, section 13.1.1, criterion III.1(a)(ii) states that the applicant should describe how the organization provides for the integrated management of activities that support the construction, operation, and maintenance of the facility.

The NRC staff reviewed PSAR section 11.1.1, which discusses the integration of organizational management at the executive level and states that the management “establishes expectations for quality, safety, and efficiency in plant operations and support activities.”

The NRC staff reviewed PSAR section 11.1.1.1, which states that the “President is responsible for all aspects of design and construction.” This section describes that design, and construction responsibilities are assigned to the executive responsible for project execution, with control of nuclear plant operation subsequently being assigned to the chief nuclear officer prior to preoperational testing. PSAR section 11.1.1.8 discusses that the quality assurance manager reports to the president and is responsible for quality assurance, quality control, and the implementation of the LME quality assurance program description (QAPD) (ML20343A088).

The NRC staff reviewed PSAR section 11.1.1.2, which states that the senior vice president of commercial operations is “responsible for the administration of reactor development, nuclear licensing, and support activities for LME under the [QAPD].” This position is responsible for ensuring that LME programs, products, and processes deliver the highest technical quality and nuclear safety for the safe, reliable, and efficient operation of nuclear plants.

The NRC staff reviewed PSAR section 11.1.1.9, which states that the “Program Construction Manager is responsible for the construction activities of the full PLM [Project Long Mott] Program and reports to the Program Director.” This position’s responsibilities include, among others, environmental, health, safety and security, quality, subcontract management, cost and schedule performance and control, as well as overall management of the physical project site and temporary facilities.

The NRC staff reviewed PSAR section 11.1.1.11, which states that the lines of authority, decision-making, and communication are clearly and unambiguously established between the project members, including contractors. Additionally, the section further states that “no later than six months prior to fuel receipt of the unit, those positions that are identified for operations will be staffed and will be given the appropriate authority required to perform operations activities.”

The NRC staff reviewed PSAR figures 11.1-2 and 11.1-3, which graphically describe the organizational structure for management of construction, operation, and maintenance activities consistent with the sections of the PSAR reviewed above.

Accordingly, because the PSAR discusses how the organization provides for the integrated management of activities that support the construction, operation, and maintenance of the

facility, the NRC staff confirmed the PSAR is consistent with the guidance in NUREG-0800, section 13.1.1, criterion III.1(a)(ii). Therefore, the NRC staff concludes that the applicant has described how the organization structure will provide for integrated management of construction, operation, and maintenance activities, consistent with criterion III.1(a)(ii) presented in NUREG-0800, section 13.1.1.

11.1.1.5.2 Obtaining Personnel

NUREG-0800, section 13.1.1, criterion III.1(a)(iii) states that the applicant should describe how it will obtain personnel with sufficient experience to provide management and technical support for the facility. The NRC staff applied this requirement towards both the construction and operational organization.

The NRC staff reviewed PSAR section 11.1, which describes that LME is leveraging a process for selection of an engineering, procurement, construction management (EPCm) contractor for the LMGS. A multifunctional team consisting of LME, X-energy, TDCC, independent nuclear construction and engineering experts, and the intended operator will conduct the contracting process, including soliciting bids, evaluating proposals, and awarding the contract.

The NRC staff reviewed PSAR section 11.1.1.7, which states that the program director reports to the president during the construction phase and is responsible for site-related construction activities. The NRC staff notes that the QAPD addresses the qualifications of personnel in technical and management roles.

Accordingly, because the PSAR describes how it will obtain personnel with sufficient experience to provide management and technical support for the facility, the NRC staff confirmed the PSAR is consistent with the guidance in NUREG-0800, section 13.1.1, criterion III.1(a)(iii) for supporting construction of the facility. The NRC staff will analyze the requirements related to the operating organization during its review of any OL application. Therefore, the NRC staff concludes that the PSAR sufficiently describes how staffing will be obtained to support construction of the facility in the CP application, and that further evaluation of acquiring personnel for the operating organization is not required and will be analyzed during the review of any OL application, consistent with criterion III.1(a)(iii) presented in NUREG-0800, section 13.1.1.

11.1.1.5.3 Control of Major Contractors

NUREG-0800, section 13.1.1, criterion III.1(a)(vii) states that the applicant should describe how the organization will carry out its responsibilities to control major contractors and has committed to consider safety first, with due consideration of risk insights, in design and construction of the facility and during the transition from construction through testing to operation.

The NRC staff reviewed PSAR section 11.1.1.2 which describes that the senior vice president of commercial operations will be responsible for ensuring quality and nuclear safety. PSAR section 11.1.1.3 describes that the environmental, health and safety (EH&S) reports to the president and has the “authority and responsibility to ensure that safety standards are incorporated in all aspects of the activities associated with LMGS.” PSAR section 11.1.1.3 also states that the “EH&S has the authority to stop work when unsafe behaviors or results are observed.”

The NRC staff reviewed PSAR section 8.1.4, which states that the LMGS reliability assurance program will be implemented using the QAP during construction phase activities. It further states that programmatic controls, including organizational controls, design controls, procedures and instructions, the corrective action process, records, audits, and expert panels, are established to assure that risk insights are identified, maintained, and communicated to the responsible organizations.

Accordingly, because the PSAR describes how the organization will carry out its responsibilities to control major contractors and has committed to consider safety first, with due consideration of risk insights, in design and construction of the facility and during the transition from construction through testing to operation, the NRC staff confirmed the applicant follows the guidance in NUREG-0800, section 13.1.1, criterion III.1(a)(vii). Therefore, the NRC staff concludes that the applicant has described how the organization will, beginning in the design and construction phases and continuing through testing to operation, control major contractors while incorporating consideration of both safety and risk, consistent with criterion III.1(a)(vii) presented in NUREG-0800, section 13.1.1.

11.1.1.5.4 Safety Responsibility

NUREG-0800, section 13.1.1, criterion III.1(b) states that the applicant should identify the corporate officer responsible for nuclear activities and should make a commitment by the applicant that this individual will have no ancillary responsibilities that might detract attention from nuclear safety matters.

The NRC staff reviewed PSAR section 11.1.1.2, which describes that the senior vice president of commercial operations is responsible for ensuring quality and nuclear safety. The NRC staff further reviewed QAPD section 1.3, which states that the senior vice president/chief nuclear officer is responsible for delivering the highest technical quality and nuclear safety by actively supporting the safety culture. Section 3.1.1 of the QAPD further notes that the QAPD provides the authority and responsibility to stop work in progress not being done where safety might be jeopardized.

Accordingly, because the PSAR identifies the corporate officer responsible for nuclear activities and safety matters, the NRC staff confirmed the PSAR is consistent with the guidance in NUREG-0800, section 13.1.1, criterion III.1(b). Therefore, the NRC staff concludes that the corporate officer responsible for nuclear activities and safety matters is identified within the application, consistent with criterion III.1(b) presented in NUREG-0800, section 13.1.1.

11.1.1.5.5 Construction Responsibilities

NUREG-0800, section 13.1.1, criterion III.1(c) states that the applicant should define the design and construction responsibilities for applicant and vendor personnel, both in terms of numbers of personnel and necessary position qualifications and experience.

The NRC staff reviewed PSAR section 11.1.1.7, which states that the program director serves as the leader in charge of the project management organization and the individual fulfilling his

role “has demonstrated experience managing and directing large and mega projects with demonstrated capacity for safety, quality, accountability and delivery of results.”

The NRC staff reviewed PSAR section 11.1.1.11, which states that the construction program management reports to the program director and is responsible for all construction activities.

The NRC staff reviewed PSAR section 11.1.1.10, which describes the role of the primary scope area construction managers, whose responsibilities include oversight of construction activities within their designated work assignments in areas including quality and technical compliance with specifications, compliance with health and safety standards, schedule adherence and documentation management for commissioning. The NRC staff notes that these positions are listed in quantity based on the number of primary scope areas, and employes a team of specialists in order to complete assigned tasks related to construction activities.

Additionally, the NRC staff reviewed PSAR section 11.1.1.9, which discusses the responsibilities of the security manager, who is responsible for access authorization and FFD programs during construction. PSAR section 11.1.1.9 also describes the responsibilities of the training manager, who is responsible “for the development, implementation, and maintenance of training to construction personnel.”

Accordingly, because the PSAR defines the design and construction responsibilities for applicant and vendor personnel, both in terms of numbers of personnel and necessary position qualifications and experience, the NRC staff confirmed the applicant follows the guidance in NUREG-0800, section 13.1.1, criterion III.1(c). Therefore, the NRC staff concludes that the PSAR adequately defines applicant and vendor responsibilities for design and construction, as well as key personnel roles and qualifications associated with construction, consistent with criterion III.1(c) presented in NUREG-0800 section 13.1.1.

11.1.1.5.6 Responsibilities for Human Factors Engineering

NUREG-0800, section 13.1.1, criterion III.1(d) states that the applicant should clearly define management and organizational responsibilities to address HFE considerations.

The NRC staff notes that PSAR section 8.7 addresses the HFE considerations for the LMGS. For NUREG-0800, section 13.1.1, criterion III.1(d), the NRC staff reviewed PSAR section 8.7.2, which states that the PLM HFE program is developed under the LME QAP. It further states that the requirements of the HFE program “flow down to suppliers and potential sub-suppliers via procurement documents.” Oversight of the suppliers and sub-suppliers is performed in accordance with the QA program.

Accordingly, because the PSAR defines management and organizational responsibilities to address HFE considerations, the NRC staff confirmed that the applicant follows the guidance in NUREG-0800, section 13.1.1, criterion III.1(d). Therefore, the NRC staff concludes that management and organizational responsibilities are clearly defined as they relate to the addressing of HFE considerations, consistent with criterion III.1(d) presented in NUREG-0800, section 13.1.1.

11.1.1.6 Plans for Conduct of Operations

DANU-ISG-2022-05, criterion 11.2(f) states that the applicant should describe plans for the conduct of operations that are consistent with the relevant portions of NUREG-0800, sections 13.5.1.1, "Administrative Procedures – General" and 13.5.2.1, "Emergency and Operating Procedures."

The NRC staff reviewed PSAR section 11.5, which describes the LMGS plan for developing, implementing, and maintaining plant procedures including: (1) administrative procedures for plant operations, (2) operating procedures, which includes normal operating, abnormal, emergency, and alarm response procedures, and (3) maintenance and other procedures, which includes test, surveillance, chemistry and radiation protection procedures. This section of the PSAR further states that the three procedures groups listed above are developed using the applicable guidance of ANSI/ANS-3.2-2012, endorsed in RG 1.33, to meet the requirements of 10 CFR Part 50, Appendix B. The PSAR also states that the procedures are to be "developed and implemented in a sufficient time frame before fuel load to permit plant staff familiarization and to facilitate regulatory review." The NRC staff notes that the applicant did not specify in PSAR section 8.7.2 the applicable principles and practices that will be used to develop the associated procedures as part of the HFE program, along with the other HFE elements found in NUREG-0711. This area is discussed in section 8.7.1 of this SE. The NRC staff therefore concludes that review of the applicant's plans for the conduct of operations is not required until the review of an OL application.

11.1.1.7 Education and Experience Requirements for Key Personnel

DANU-ISG-2022-05, criterion 11.2(g) states that the applicant should propose education and experience requirements for key personnel that conform to the qualifications and experience guidance endorsed by RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants." The NRC staff notes that RG 1.8 endorses ANSI/ANS 3.1, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants."

The NRC staff reviewed PSAR section 11.3.2, which states that the licensed operator training program will include requirements for minimum education and experience of trainees, as determined in coordination of the systems approach to training (SAT) program development. The PSAR further states that these requirements will be submitted as part of an OL application for approval as part of the training program for licensed operators and will "provide reasonable assurance that licensed operators are trained and qualified… to operate and maintain the plant in a safe manner."

The NRC staff reviewed PSAR section 11.3.3, which describes the training programs for non-licensed personnel. The PSAR states that RG 1.8 and ANSI/ANS-3.1, with exceptions and clarifications, will be reviewed for applicability and suitability for LMGS plant staff positions when determining education and experience requirements. The PSAR describes that development of the minimum requirements will be determined systematically in accordance with SAT training program development. The PSAR further states that these minimum requirements will "provide reasonable assurance that licensed operators are trained and qualified with sufficient knowledge and skill to operate and maintain the plant in a safe manner during all modes and

conditions of operation." These requirements will be submitted for approval as part of the training programs for non-licensed operators.

The NRC staff therefore concludes that the PSAR addresses education and experience requirements for key personnel that are consistent with the guidance of RG 1.8, and consistent with criterion 11.2(g) presented in DANU-ISG-2022-05.

11.1.1.8 Proposed Operating Organization

DANU-ISG-2022-05, criterion 11.2(h) states that an applicant's proposed organization should conform to the guidance of RG 1.33 or the applicant should provide justification for an alternative. The NRC staff notes that RG 1.33 endorses ANSI/ANS 3.2.

The NRC staff reviewed PSAR section 11.1.2.3, which provides a description of the duties of the plant general manager. Included in those responsibilities are the development of the site operating and support staff, and operation of the new nuclear plant during all test phases. During the transition to operating phase activities, this position "assures the safe, reliable, and efficient operation of the plant within the constraints of applicable regulatory requirements, the operating license, and the QAP."

The NRC staff reviewed PSAR section 12.1.1.4, which states that the LMGS organizational arrangement allows a division of resources into clear construction, commissioning, and operations teams for each module. PSAR section 12.1.1.6 describes the turnover process from construction to cold commissioning, with formal turnover packages of all construction activities that include inspections, analyses, and acceptance or performance criteria.

The NRC staff notes that the PSAR does not describe whether the operating organization will comply with the organizational provisions of ANSI/ANS 3.2 or with an alternative approach. However, given that this criterion specifically addresses the facility organization during the operating phase, the NRC staff concludes that the evaluation of this criterion, including any related commitment to ANSI/ANS 3.2, is not required until the review of an OL application.

11.1.1.9 Compliance with Minimum Staffing Requirements

DANU-ISG-2022-05, criterion 11.2(i) states that an applicant should describe how an adequate number of licensed operators will be available at all required times to satisfy the minimum staffing requirements of 10 CFR 50.54(m), or else provide justification for an exemption.

The NRC staff reviewed PSAR section 11.2.3 which states that LME plans to perform a staffing optimization analysis, using the HFE program developed in accordance with NUREG-0711, "Human Factors Engineering Program Review Model", Revision 3 (ML12324A013), to determine the number of licensed operators as well as the qualifications of control room and supporting personnel. The PSAR further states that LME anticipates that the final staffing analysis is expected to confirm an optimum control room staffing level of three licensed operators (i.e., one shift supervisor and two control room operators) controlling four reactor modules from a single control room using the guidance described in "Xe-100 Licensing Topical Report Control Room Staffing Analysis Methodology" (ML23258A243). PSAR section 11.2.3 acknowledges that the staffing level and configuration does not meet the requirements of 10 CFR 50.54(m). However, the NRC staff notes that PSAR section 11.2.3 states that a staffing

optimization analysis will be conducted consistent with NUREG-1791, "Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)" (ML052080125), and that the applicant intends to provide sufficient justification for an exemption of 10 CFR 50.54(m), which would be submitted as part of an OL application.

Accordingly, because the PSAR defines how LME plans to perform a staffing optimization analysis, using the HFE program, to determine an adequate number of licensed operators to comply with 10 CFR 50.54(m) and NUREG-1791, the NRC staff confirmed that the applicant follows the guidance in DANU-ISG-2022-05, criterion 11.2(i). Therefore, the NRC staff concludes that the applicant has described preliminary plans for licensed operator staffing to the extent warranted for a PSAR consistent with criterion 11.2(i) presented in DANU-ISG-2022-05, and that further review of compliance with the staffing requirements at the OL stage, including the planned exemption requests of 10 CFR 50.54(m) and NUREG-1791 is not required until the review of an OL application.

11.1.1.10 Engineering Expertise on Shift

DANU-ISG-2022-05, criterion 11.2(j) states that the applicant should describe how engineering expertise on shift will be consistent with the Commission Policy Statement on Engineering Expertise on Shift (50 FR 43621) and within the guidelines of Three-Mile Island (TMI) Action Plan Item I.A.1.1, "Shift Technical Advisor," of NUREG-0737, "Clarification of TMI Action Plan Requirements" (November 1980; ML051400209), or else provide justification for an alternative.

The NRC staff reviewed PSAR section 11.3.3, which states that the shift technical advisor (STA) role "will be evaluated to determine if the role is necessary for the safe operation of the LMGS during emergency conditions." The PSAR further states that the guidance of SECY-21-0039, "Elimination of the Shift Technical Advisor for the NuScale Design," (ML21060A823) will be considered and, if necessary, an exemption request with justification of the exemption determination will be submitted as part of any future OL application.

The NRC staff therefore concludes that the technical justification for the training or omission of the STA is not required until the OL stage to ensure compliance with 10 CFR 50.120(b)(2)(iii) or 10 CFR 50.12.

11.1.1.11 Role and Function of Vendors During Design and Construction

DANU-ISG-2022-05, criterion 11.2(k) states that the applicant should describe the role and function of the architect-engineer and the nuclear steam supply system vendors during design and construction in a manner consistent with NUREG-0800, section 13.1.1, as well as the organizational controls over the project-related activities of the architect-engineer and nuclear reactor vendors, including preservation of documentation.

The NRC staff reviewed PSAR section 11.1.1.8, which states that the quality assurance manager is responsible for quality assurance, quality control, and the implementation of the LME QAPD. This includes, in part, independent oversight of quality activities, and ensuring that vendors providing quality services, parts, and materials to LME are meeting the requirements of

10 CFR 50, Appendix B, through audits. The NRC staff further reviewed PSAR section 11.1.1.7, which states that the program director is responsible for site-related construction activities.

The NRC staff notes that PSAR section 11.1.1.9 discusses the program construction manager, who is responsible, in part, for the quality, subcontract management, and overall management of the project. The PSAR states that there are construction managers for each primary scope area who report to the construction manager, and PSAR section 11.1.1.10 describes their responsibilities to include leading a team of specialists that ensure the contractor delivers quality and technical compliance with specification, performs all work in a safe manner, and maintains and updates all documentation necessary for smooth turnover to commissioning.

Additionally, the NRC staff notes that QAPD section 17.1, “Record Retention,” states that records of activities related to “design, engineering, procurement, manufacturing, inspection, test, and audits and their retention times” are retained in accordance with Regulatory Position C.3.a of RG 1.28, “Quality Assurance Program Criteria (Design and Construction),” Revision 5.

Accordingly, because the PSAR described the role and function of the architect-engineer and the nuclear steam supply system vendors during design and construction in a manner consistent with NUREG-0800, section 13.1.1, as well as the organizational controls over the project-related activities of the architect-engineer and nuclear reactor vendors, including preservation of documentation, the NRC staff confirmed the applicant follows the guidance in DANU-ISG-2022-05, criterion 11.2(k). Therefore, the NRC staff concludes that the applicant has described the roles of the architect-engineer and the nuclear steam supply system vendors, organizational controls over project-related activities, and provisions for the preservation of documentation, consistent with criterion 11.2(k) presented in DANU-ISG-2022-05.

11.1.1.12 Reporting Responsibilities and Authorities

DANU-ISG-2022-05, criterion 11.2(l) states that the applicant should identify and describe the reporting responsibilities and authorities in the functional areas of radiation protection/health physics, quality assurance, and training. Additionally, this criterion states that the associated reporting responsibilities and authorities should ensure independence from normal operating pressures.

The NRC staff reviewed PSAR section 11.1.2.3, which states that the plant general manager has overall responsibility for occupational and public radiation safety, with responsibilities consistent with the guidance in RG 8.8 “Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable (ALARA)” and RG 8.10 “Operating Philosophy for ALARA.”

The NRC staff reviewed PSAR section 11.1.2.5, which states that the general manager of training and organizational effectiveness will have a similar structure to the training organization of the NextEra Energy fleet. This position has the responsibility for the training of the plant staff and reports to the fleet vice president – nuclear.

The NRC staff reviewed PSAR section 11.1.2.7, which states the quality assurance and control inspector responsibilities include oversight of safety-related activities associated with the operation of the plant. The section further states that “the site QA organization is independent of

the unit operating line organization." Specifically, the NRC staff notes that PSAR section 11.1.2.7 states that the "QA and QC [quality control] personnel report to the Site Quality Manager."

Accordingly, because the PSAR described the reporting responsibilities and authorities in the functional areas of radiation protection/health physics, quality assurance, and training, including independence from normal operating pressures, the NRC staff confirmed the PSAR is consistent with the guidance in DANU-ISG-2022-05, criterion 11.2(l). Therefore, the NRC staff concludes that the applicant has adequately described reporting responsibilities and authorities for the areas of radiation protection, quality assurance, and training and, furthermore, has presented an organizational reporting structure that should reasonably provide independence from normal operating pressures, consistent with criterion 11.2(l) presented in DANU-ISG-2022-05.

11.1.1.13 Responsibilities of the Operating Organization

DANU-ISG-2022-05, criterion 11.2(m) states that the applicant should define the responsibilities of the operating organization related to activities important to the safe operation and maintenance of the facility. Functional areas should be separately supervised and managed.

The NRC staff reviewed PSAR section 11.1.2.4, which describes the duties of the operations coordinator. This position has the overall responsibility for the daily operation of the assigned units. PSAR section 11.1.2.6 describes the responsibilities of the maintenance coordinator, including the performance of preventative and corrective maintenance, including directing activities that modify the station in accordance with applicable standards. The NRC staff notes that both the operations coordinator and maintenance coordinator positions report to the plant general manager, but serve as separate, distinct positions providing supervision and direction to their respective functional area.

Accordingly, because the PSAR describes the responsibilities of the operating organization related to activities important to the safe operation and maintenance of the facility, with functional areas separately supervised and managed, the NRC staff confirmed that the applicant follows DANU-ISG-2022-05, criterion 11.2(m). Therefore, the NRC staff concludes that the applicant has defined the responsibilities of the operating organization as they relate to facility operation and maintenance, with key organizational functions being separately supervised and managed, consistent with criterion 11.2(m) presented in DANU-ISG-2022-05.

11.1.1.14 Managerial Depth

DANU-ISG-2022-05, criterion 11.2(n) states that in order to address 10 CFR 50.34(b)(6)(v), the applicant should describe how sufficient managerial depth is available to provide qualified backup, consistent with the QAP and ANS 3.2, for overall station operation in the event of temporary unexpected contingencies.

The NRC staff noted that the QAPD section 2.5, "Personnel Training and Qualifications," states that "personnel assigned to implement elements of the QAPD shall be capable of performing their assigned tasks." This is accomplished through indoctrination, training, and qualification, as

necessary. The QAPD further states that sufficient managerial depth is provided to cover absences of incumbents.

The NRC staff notes that ANSI/ANS-3.2 addresses the operational phase of plant life, which is entered prior to initial fuel loading. As such, the NRC staff determined that evaluation of the consistency of the operating organization with either ANSI/ANS-3.2 or with a proposed, alternative approach is not required until the review of an OL application.

The NRC staff therefore concludes that the applicant has described how sufficient managerial depth will be available to provide qualified backup for overall station operation to the extent warranted for a PSAR, consistent with criterion 11.2(n) presented in DANU-ISG-2022-05, and that an additional review of this area is not required until the review of an OL application to ensure compliance with 10 CFR 50.34(b)(6)(v).

11.1.1.15 Plans for Preventing Personnel Impairment from Fatigue

DANU-ISG-2022-05, criterion 11.2(o) states that in order to address 10 CFR 26.205(c), the applicant should describe how the numbers of licensed and non-licensed personnel subject to 10 CFR 26.205 will be sufficient to allow shift schedules that prevent personnel impairment from fatigue due to the duration, frequency, or sequencing of successive shifts. The NRC staff notes that 10 CFR 26.3(c), requires, in part, that CP holders shall comply with the requirements of 10 CFR Part 26, Subpart I, “Managing Fatigue,” by no later than the receipt of special nuclear material in the form of fuel assemblies.

PSAR section 11.6.2 states that the construction FFD program remains in effect until the construction site transitions to a phase requiring site-wide implementation of 10 CFR Part 26, Subpart I. The PSAR further states that the FFD program, including the schedule for its implementation, will be described in the final safety analysis report (FSAR) for an OL application.

The NRC staff therefore concludes that the applicant has, to an extent appropriate for a PSAR, described how the numbers of personnel subject to 10 CFR 26.205 will be sufficient to prevent personnel impairment from fatigue, consistent with criterion 11.2(o) presented in DANU-ISG-2022-05, and that an additional review of this area is not required until the review of an OL application to ensure compliance with 10 CFR 26.205(c).

11.1.1.16 Training Program for Licensed Operators

DANU-ISG-2022-05, criterion 11.2(p) states that the applicant should describe a training program for licensed operators that is consistent with the requirements of 10 CFR Part 55.

The NRC staff reviewed PSAR section 11.3.2, which states that the operator training program will be developed and implemented in accordance with 10 CFR Part 55 and that a description of the operator re-qualification program will be provided at the OL application stage. The NRC staff notes that PSAR section 11.3.2 refers to submitting exemptions to some of the requirements for 10 CFR Part 55, including portions of medical requirements, application requirements in NRC Form 398, and operator examination requirements using NUREG-1021 but did not provide technical basis for the potential exemptions. The NRC staff therefore finds that the requirements

of 10 CFR Part 55 are applicable to the LMGS and would review any exemptions submitted in an OL application on a case-by-case basis.

The NRC staff therefore concludes that the applicant has described the licensed operator training program to the extent warranted for a PSAR, consistent with criterion 11.2(p) presented in DANU-ISG-2022-05, and that further review of the initial and continuing licensed operator programs is not required until the review of an OL application.

Additionally, the NRC staff notes that PSAR section 11.3.2 states that the licensed operator training program will use a simulation facility approved by the Commission under 10 CFR 55.46(b), and will conform with RG 1.149, "Nuclear Power Plant Simulation Facilities for Use in Operator Training, License Examinations, and Applicant Experience Requirements," which endorses ANSI/ANS-3.5, "Nuclear Power Plant Simulators for Use in Operator Training and Examination," with clarifications. The NRC staff further notes that 10 CFR 50.34(f)(2)(i) requires a construction permit application to provide sufficient information to demonstrate that simulator related requirements will be satisfactorily completed by the operating license stage, including the provision of simulator capability that accurately models the control room and can simulate small-break LOCAs. PSAR section 11.3.2 also states that a request for Commission approval of the simulation facility will be submitted in the operating license application preparation phase.

Therefore, because the PSAR commits to developing and obtaining approval of a simulation facility in accordance with 10 CFR 55.46(b), the NRC staff finds that the application provides sufficient information to demonstrate that simulator requirements will be addressed at the OL stage. Accordingly, the NRC staff concludes that the simulation facility description is acceptable for a CP application and that further review is not required until the review of an OL application.

11.1.1.17 Training Program for Non-Licensed Personnel

DANU-ISG-2022-05, criterion 11.2(q) states that the applicant should describe a training program for non-licensed nuclear plant personnel that meets the requirements of 10 CFR 50.120(b)(2) and (3).

The NRC staff reviewed PSAR section 11.3.1.1, which states that the training programs will be developed, implemented, and maintained using SAT as defined in 10 CFR 55.4. The NRC staff notes that the section includes provisions for the training and qualification of categories of nuclear power plant personnel listed in 10 CFR 50.120(b)(2); the PSAR includes a table that correlates those positions into positions used at LMGS. The NRC staff further notes that PSAR section 11.3.3 discusses the intent to submit an exemption request as part of an OL application for the STA position. The STA position is required by 10 CFR 50.120(b)(2)(iii).

The NRC staff reviewed PSAR section 11.3.1, which states that the training programs will include applicable criterion as required by 10 CFR 50.120(b)(3). Included in the list are measures for periodic evaluation and revision, management reviews, and record retention. The NRC staff further notes that PSAR sections 11.3.2 and 11.3.3 affirm that the training programs will include the necessary materials to allow qualified personnel to operate and maintain the plant in a safe manner in all modes of operation. PSAR section 11.3.1 further describes the scope of the training programs utilizing job task analysis, and includes, in part, the full spectrum of plant functions and systems, full range of relevant HSIs, and the full extent of plant conditions. Based on the above, the NRC staff determined that the PSAR adequately describes the training program for non-licensed nuclear plant personnel that meets the applicable

requirements of 10 CFR 50.120(b)(2) and (b)(3) and will be approved and ready to administer no later than 18 months prior to initial fuel load.

The NRC staff therefore concludes that the applicant has described plans for establishing training programs for non-licensed nuclear plant personnel to the extent appropriate for a PSAR, consistent with criterion 11.2(q) presented in DANU-ISG-2022-05, and that these programs will be evaluated, approved, and ready to administer no later than 18 months prior to initial fuel load to ensure compliance with 10 CR 50.120(b)(2) and (3).

11.1.2 Conclusion

Based upon the evaluations described in section 11.1.1 of this SE, and since the applicant has described plans for the design, construction, and operating organization that are appropriate for a construction permit, the NRC staff finds that 10 CFR 50.34(a)(6), 50.34(a)(9), 26.3(c), 50.35, "Issuance of construction permits," and 50.40 for the issuance of a CP are met because the associated criteria of DANU-ISG-2022-05 and NUREG-0800 section 13.1.1 have been satisfied to the extent warranted for a PSAR.

11.2 Concept of Organization

Concept of organization for the LMGS was evaluated in section 11.1 of this SE.

11.3 Training

Training for the LMGS was evaluated in section 11.1 of this SE.

11.4 Emergency Planning

This section of the SE describes the NRC staff's review and evaluation of the LMGS PSAR section 11.4, which provides a description of the LMGS facility preliminary plans for coping with emergencies as required by 10 CFR 50.34(a)(10) to be included in the CP application PSAR. The LMGS facility emergency preparedness preliminary plans are based on the alternative, performance-based requirements in 10 CFR 50.160, ''Emergency preparedness for small modular reactors, non-light-water reactors, and non-power production or utilization facilities." PSAR table 1.4-2 identifies that the design will conform to the guidance in RG 1.242, "Performance Based Emergency Preparedness for Small Modular Reactors" (ML23226A036). The LMGS site emergency response preliminary plans for coping with emergencies will be further developed into a final LMGS site performance-based emergency plan and submitted to the NRC for review as part of an OL application submittal. PSAR section 11.4 states that the emergency plan will be developed to meet the requirements in 10 CFR 50.160, using the guidance in RG 1.242.

<u>Staff's Emergency Preparedness SE Layout and Structure</u>

Because the performance-based emergency response requirements in 10 CFR 50.160(b)(1) are, at times, interrelated, several regulatory criteria overlap with similar technical subject matter design details. The following SE sections organize the NRC staff's review by common emergency response technical subject matter where appropriate, thus leading to several 10 CFR 50.160(b)(1) criteria being reviewed under the same SE subsection. Where an emergency response technical discussion supports more than one provision of 10 CFR 50.160,

the SE identifies the applicable requirements and provides the NRC staff's findings for each requirement based on the integrated evaluation.

The applicable regulatory requirements for the preliminary plans for coping with emergencies are as follows:

- 10 CFR 50.33(g)(2)
- 10 CFR 50.34(a)(10)
- 10 CFR 50.34(a)(10)(ii)
- 10 CFR 50.160(b)

The applicable guidance that the NRC staff used to evaluate preliminary emergency planning design information is as follows:

- DANU-ISG-2022-01, Appendix C, "Construction Permit Guidance," section titled "Detailed Non-Light Water Reactor Construction Permit Guidance," provides NRC staff guidance for CP applicants that have not supplied all of the technical information required to support the issuance of a CP, and addresses the minimum information necessary in a CP application for the NRC staff to make the findings for a CP under 10 CFR 50.35(a). (ML23277A139).
- NUREG-0800, "Standard Review Plan," section 13.3, "Emergency Planning," section III, "Review Procedures," subsection "General Review Procedures," guidance item 3, provides guidance on the extent of a CP review, and section IV, "Evaluation Findings" guidance item 1, "Construction Permit," provides NRC staff evaluation guidance on a CP preliminary plan's acceptability and the PSAR contains sufficient information for NRC staff findings and conclusions (ML063410307).
- RG 1.242, "Performance Based Emergency Preparedness for Small Modular Reactors" (ML23226A036).

11.4.1 Technical Evaluation

11.4.1.1 The LMGS Site, Plume Exposure Pathway (PEP) Emergency Planning Zone (EPZ) Boundary and EPZ Physical Characteristics, and PEP EPZ Size Determination Analysis Evaluation

The following SE subsections document the NRC staff's review of the PEP EPZ boundary and physical characteristics, as well as the applicant's preliminary sizing analysis described in the LMGS CP application to determine the size of the LMGS site PEP EPZ.

11.4.1.1.1 LMGS Site Layout and PEP EPZ Boundary and Physical Characteristics General Description Evaluation (10 CFR 50.160(b)(3))

PSAR chapter 1, "General Plant and Site Description and Overview of Licensing Basis," states that the proposed LMGS facility will be constructed on land owned by TDCC, in Calhoun County, Texas (TX), approximately eight miles (12.9 km) north-northwest of Seadrift, TX. The proposed LMGS facility will be constructed adjacent to Dow Seadrift Operations (SDO) and will be located within 0.5 mi (0.8 km) of Dow Seadrift Operations at their nearest points. The proposed LMGS site will be bounded on the north, east, south, and west by land that is owned by TDCC. PSAR figure 2.1-3 displays the LMGS facility site center point and site location relative to the TDCC facility. PSAR section 2.1.1.1 states that the LMGS site boundary coincides with the LMGS exclusion area boundary (EAB). The EAB is 400 meters from the LMGS facility radiological release envelope. The radiological release envelope, a rectangle circumscribing the reactor building, the fuel handling annex building, and the helium service facility is shown in PSAR figure 1.1-3.

The LMGS site and EAB reside entirely within the property owned by TDCC. PSAR section 2.1.2 states that there are no public roads, railroads, or waterways that traverse the LMGS EAB, nor are there any residences within the EAB. The section also describes that the LMGS operator, NEER, will have ultimate authority for controlling all activities within the LMGS site EAB, including exclusion and evacuation of personnel and removal of property from the EAB. PSAR figure 2.1.2-1 displays the LMGS site EAB, proposed site access roads, and the owner-controlled property boundary.

The NRC staff reviewed Technical Report number 010810, "Preliminary Plume Exposure Pathway Emergency Planning Zone Analysis," revision 4, section 6, "Conclusion," which states that the proposed preliminary PEP EPZ is established at the EAB (400 m) for the LMGS. The PSAR lists Technical Report number 010810 (ML25090A066) in PSAR table 1.4-4, which incorporates the technical report by reference into the PSAR. PSAR figure 1.1-3, note 2 states that the PEP EPZ distance is the same as the EAB.

Based on the NRC staff's review and evaluation of the LMGS proposed site and PEP EPZ boundary, documented in this SE section, the NRC staff determined that the PSAR preliminary plans describing the LMGS facility site and PEP EPZ boundary contain sufficient RG 1.242 emergency response preliminary plan information to address the guidance described in DANU-ISG-2022-01. Therefore, the NRC staff finds that the LMGS proposed site and preliminary PEP EPZ boundary PSAR design descriptions meet the regulatory requirements of 10 CFR 50.34(a)(10) with respect to the PSAR content and meet the regulatory requirements of 10 CFR 50.34(a)(10)(ii) because LME has provided a discussion of the preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 as evaluated in SE section 11.4.

11.4.1.1.2 LMGS Facility PEP EPZ Size Analysis Determination Evaluation (10 CFR 50.33(g)(2)(i))

The NRC staff reviewed PSAR section 11.4.2.1, which states that the preliminary PEP EPZ analysis is documented in X-energy technical report, "Preliminary Plume Exposure Pathway Emergency Planning Zone Analysis," (hereafter referred to as preliminary PEP EPZ analysis; ML26261A428). The entire preliminary PEP EPZ analysis is incorporated by reference into the PSAR and is listed in PSAR table 1.4-4.

The preliminary PEP EPZ analysis, section 6, states that the LMGS site PEP EPZ is established at the EAB (400 m). Preliminary PEP EPZ analysis, section 1.1, "Purpose," states, in part, that to establish the site-specific PEP EPZ size for the LMGS site, the PEP EPZ sizing methodology in X-energy Licensing Topical Report (TR), "Plume Exposure Pathway Emergency Planning Zone Sizing Methodology" (hereafter referred to as PEP EPZ methodology; ML26075E856) is applied. The PEP EPZ methodology, Revision 1 (ML24331A308), was submitted for NRC staff review and approval on November 26, 2024. The NRC staff approved the PEP EPZ methodology, Revision 2 (ML26075E859), on February 11, 2026 (ML26034C377).

The PEP EPZ methodology is a risk-informed approach consistent with the guidance in RG 1.242, Appendix A, "General Methodology for Establishing Plume Exposure Pathway Emergency Planning Zone Size," for determining PEP EPZ size to meet the requirements in 10 CFR 50.33(g)(2) and 10 CFR 50.160(b)(3). Both the PEP EPZ methodology and the preliminary PEP EPZ analysis follow the guidance of RG 1.242. The preliminary PEP EPZ analysis provides results necessary to compare to the PEP EPZ sizing criteria in 10 CFR 50.33(g)(2)(i).

Preliminary PEP EPZ analysis, section 1.1, states that the PEP EPZ methodology provides "an approach for determining the [LMGS PEP EPZ] size based on the area within which public dose, as defined in 10 CFR 20.1003, 'Definitions,' is projected to exceed 10 millisievert (mSv) (1 rem) total effective dose equivalent over 4 days (96 hours) from the release of radioactive materials from the facility, considering accident likelihood and source term, timing of the release sequence, and meteorology. Additionally, the methodology addresses the area in which predetermined, prompt protective measures are necessary."

The NRC-approved PEP EPZ methodology included seven limitations and conditions (L&Cs). Specifically, the NRC staff's SE for the PEP EPZ methodology imposed the following L&Cs:

1. The probabilistic risk assessment (PRA) used to implement the PEP EPZ methodology are design- and site-specific and the PRA was developed for all applicable hazards, all plant operating states, and all sources of radioactive material, using the guidance in RG 1.247, "Acceptability of Probabilistic Risk Assessment Results for Non-Light-Water Reactor Risk-Informed Activities" (ML21235A008) and Appendix A to 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). Prior to the initial fuel loading, any exceptions to meeting capability categories referred to in RG 1.247, must be justified and documented.

2. An applicant that references the PEP EPZ methodology must justify the technical acceptability of the PRA(s) performed for the selected hazards and plant operating states (e.g., site-specific scoping level seismic PRA). Prior to the initial fuel loading, PRAs supporting the PEP EPZ methodology must include all applicable hazards and plant operating states.

3. An applicant that references the PEP EPZ methodology must provide discussions of: (1) how PRA key assumptions and key sources of uncertainty for each analyzed hazard, plant operating state, and radioactive source were identified; (2) how the key assumptions and key sources of uncertainty identified as having the potential to significantly impact the PRA results have been characterized in a manner which is consistent with the current state of knowledge; and (3) how the impact of each identified key assumption and source of uncertainty was assessed and dispositioned.

4. An applicant that references the PEP EPZ methodology and exercises the methodology in Topical Report 010229 section 3.6, “Selection of Seismic LBEs,” and section 6.3, “Assessment for Seismic Hazards,” to choose the seismic scenario for the EPZ sizing analysis must provide site-specific justification for the use of the **[[ ]]**.

5. A condition of the use of atmospheric dispersion assumptions in the PEP EPZ methodology is that an applicant for a specific site referencing the PEP EPZ methodology must perform for the final PEP EPZ size, a dispersion analysis with site-specific data to either: (1) affirm that the results using these atmospheric dispersion assumptions are conservatively bounding, or (2) provide site-specific analysis results using dispersion factors developed during the site characterization.

6. A periodic evaluation of the PEP EPZ sizing analysis must be performed following an update or upgrade to the user’s PRAs based on a review of changes to the plant structures, systems, and components, operational practices, and applicable plant and industry operational experience. Any changes to the emergency preparedness plan resulting from the periodic evaluation should be conducted under 10 CFR 50.54(q).

7. The PEP EPZ methodology is limited to applicants that have design bases that are developed in accordance with the guidance in NEI 18-04, as endorsed by RG 1.233.

The NRC staff evaluated the LMGS PSAR to determine whether the seven L&Cs on the use of the PEP EPZ methodology were either met or reviewed as part of the PSAR and the preliminary PEP EPZ analysis. The NRC staff reviewed and evaluated the applicant’s use of the PEP EPZ methodology in the preliminary PEP EPZ analysis and determined that the preliminary PEP EPZ analysis includes discussion of the use of the LMGS PRA in the PEP EPZ sizing analysis. This same PRA was used in the licensing modernization project-based PSAR analysis. The NRC staff’s review of the LMGS PRA is discussed in section 3.1.1 of this SE. Therefore, based on NRC staff’s evaluation of the CP stage PRA as discussed in section 3.1.1 of this SE, the NRC staff finds that the preliminary PEP EPZ analysis addresses L&Cs 1 through 3 and 7 of the PEP

EPZ methodology and addresses the minimum information necessary in a CP application as described in DANU-ISG-2022-01.

However, the NRC staff notes that, consistent with the PEP EPZ methodology, because the PEP EPZ methodology's **[[ ]]** does not govern the determination of the EPZ earthquake, the NRC staff determined that L&C 4 did not apply to this review. In addition, the NRC staff notes that L&Cs 5 and 6 are not applicable to the preliminary PEP EPZ size determination in the CP application because L&Cs 5 and 6 are applicable for the final PEP EPZ size determination.

The NRC staff reviewed the preliminary PEP EPZ analysis in comparison to the PEP EPZ methodology. Because the PEP EPZ methodology was approved after the preliminary PEP EPZ analysis was submitted for NRC staff review, the NRC staff's review focused on the differences between these two reports to assess whether the methods used in the preliminary PEP EPZ analysis were acceptable.

Event Selection

The preliminary PEP EPZ analysis uses the LMGS design- and site-specific PRA to evaluate the consequences of a range of events. The preliminary PEP EPZ analysis, section 4.2, "Assessment of Seismic Hazards," states that **[[**

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However, the analysis did assess the consequences to bound a seismic scenario consistent with the PEP EPZ methodology. The NRC staff determined that the scope of the PRA (i.e., internal events) supports the CP application stage as discussed in section 3.1.1 of this SE and, therefore, the NRC staff also determined that the use of the CP stage PRA in the preliminary determination of the PEP EPZ size is acceptable. Based on the preceding discussion, the NRC staff determined that the use of the PRA to develop the range of events and the use of a seismic scenario is consistent with PEP EPZ methodology and therefore determined that the PEP EPZ analysis has considered an acceptable range of potential events.

Preliminary PEP EPZ analysis, section 5, describes the preliminary radiological consequence analysis results. The NRC staff determined through review of the PEP EPZ size analysis descriptions in comparison to the licensing basis event (LBE) analysis in PSAR chapter 3, that the preliminary PEP EPZ analysis covers a full spectrum of likely accident/event scenarios for the proposed LMGS Xe-100 reactor. The preliminary PEP EPZ analysis also evaluated the dose consequences from potential seismic events. As stated in the preliminary PEP EPZ analysis, **[[**

]] The NRC staff identified that the preliminary PEP EPZ analysis is sufficient to represent seismic scenarios as determined with relation to the preliminary LMGS design information for the CP application. Specifically, the NRC staff determined that it is reasonable to consider damage from seismic scenarios in the preliminary PEP EPZ analysis for the CP application.

PEP EPZ methodology, section 3.4.3 states that security-related events are adequately considered during the hazards screening process for determining the PEP EPZ. The section also provides that the qualitative or quantitative assessment will be performed and documented as part of the PEP EPZ sizing calculation. Preliminary PEP EPZ analysis, section 4.2 describes

a qualitative assessment of security-related events, including the inherent safety and design attributes of the Xe-100 plant and security by design features relevant to security event mitigation. The NRC staff determined that the qualitative assessment of security-related events provided in the preliminary PEP EPZ analysis, in addition to the comprehensive assessment of the site-specific LBEs is consistent with the guidance in RG 1.242, Appendix A, NRC staff position A-3.1 and is acceptable because the applicant determined the radiological releases from the facility that are evaluated in the radiological dose assessment to aid in the determination of the PEP EPZ and sufficiently described the LBEs relevant to the facility.

Based on the NRC staff's review as discussed above, the NRC staff concludes that the event selection for the preliminary PEP EPZ analysis is acceptable because it is consistent with the PEP EPZ methodology and conforms with the guidance in RG 1.242, Appendix A.

Atmospheric Dispersion

The preliminary PEP EPZ analysis presents an atmospheric dispersion analysis to support the dose calculations for the PEP EPZ analysis. The proposed methodology, which is referenced in sections 3.2 and 5.2 is explained in detail in Appendix B of the preliminary PEP EPZ analysis. The applicant uses the dispersion model ARCON with meteorological data taken from the Corpus Christi International airport and atmospheric stability determined using the Turner method to perform the dispersion calculations. The methodology for calculating the mean X/Q values used in the preliminary PEP EPZ analysis are based on the methodology outlined in RG 1.249.

The scope of the NRC staff's review was limited to assessing the proposed methodology used to perform the calculations because the preliminary analysis of the atmospheric dispersion factors is based on meteorological data from a local airport. The applicant states in Appendix B that it is not requesting a full approval for this EPZ dispersion methodology as this is a preliminary analysis while the onsite meteorological data are not yet available, therefore the NRC staff did not assess the preliminary meteorological data used. As such, the data set is not approved for use in additional licensing actions without approval by the NRC staff. Based on the NRC staff's review, the dispersion methodology was deemed reasonable because the applicant based the methodology outlined in RG 1.249. Therefore, the NRC staff concluded that the proposed methodology and values are reasonable for use in this preliminary PEP EPZ analysis.

Preliminary PEP EPZ Sizing Consequence Analysis

The NRC staff reviewed the preliminary PEP EPZ analysis, sections 4.1 and 4.2, which use the PEP EPZ methodology to determine the consequences for the spectrum of events in the PEP EPZ analysis and assesses the analysis against the PEP EPZ sizing criteria. The NRC staff's review of the provided information in the preliminary PEP EPZ analysis confirmed that the description of the events that are LBEs and the source term methodology are the same as that described in PSAR chapter 3. The NRC staff's evaluation of the LBE source terms is discussed in section 3.1.4 of this SE. The NRC staff also confirmed that the determination of source terms for the seismic scenario followed the same methodology used for the LBEs. The preliminary PEP EPZ analysis used the approved PEP EPZ methodology dose-related criteria to determine the size of the PEP EPZ in accordance with the requirements in 10 CFR 50.33(g)(2)(i).

In section 5.3 of the preliminary PEP EPZ analysis, the applicant used the PEP EPZ methodology to develop the associated dose-versus-distance curves for each LBE and assessed the potential consequences associated with the spectrum of LBEs. The preliminary PEP EPZ analysis results show that, for all evaluated events, doses may exceed the PEP EPZ sizing dose criteria within the EAB, and no doses exceeded the PEP EPZ sizing dose criteria at or beyond the EAB. Therefore, the NRC staff determined that, consistent with the approved PEP EPZ methodology, the applicant assessed and found that protective measures are warranted for onsite personnel within the EAB. Also consistent with the PEP EPZ methodology, the NRC staff confirmed that the applicant established the PEP EPZ size at the EAB.

The NRC staff's review determined that the preliminary PEP EPZ analysis was performed consistent with the PEP EPZ methodology. Therefore, the NRC staff concluded that the consequence analysis is consistent with the approved PEP EPZ methodology and supports the preliminary establishment of a PEP EPZ size set at the LMGS site EAB because it follows the guidance in RG 1.242, Appendix A.

Staff's Findings and Conclusions of the LMGS Facility PEP EPZ Size Determination Analysis Evaluation (10 CFR 50.33(g)(2)(i), 10 CFR 50.160(b)(3))

The NRC staff's determination in section 11.4.1.1.2 of this SE: (1) supports a preliminary finding that the LMGS site PEP EPZ can be set at the LMGS site EAB, (2) addresses the minimum information necessary in a CP application as described in DANU-ISG-2022-01, and (3) conforms with the PEP EPZ sizing methodology guidance in RG 1.242, Appendix A. Therefore, the NRC staff determined that the preliminary LMGS site PEP EPZ size determination analysis information is acceptable and meets the applicable PSAR preliminary plan discussion requirement for a CP application in 10 CFR 50.34(a)(10)(ii), identifies the PEP EPZ area in 10 CFR 50.33(g)(2)(i), and the PEP EPZ boundary and physical characteristics description requirements of 10 CFR 50.160(b)(3).

The final LMGS PEP EPZ emergency plan design information will be evaluated by the NRC staff at the OL stage, should the applicant apply for an OL. The OL application would be required under 10 CFR 50.34(b)(6)(v) to include the LMGS FSAR and the final LMGS site emergency plan.

11.4.1.2 LMGS Performance-Based Emergency Preparedness Framework (10 CFR 50.160(b)(1))

In accordance with 10 CFR 50.160(b)(1), applicants are required to demonstrate effective response in drills and exercises for emergency and accident conditions. Applicants with an established site boundary PEP EPZ must have an emergency plan that meets the performance-based requirements framework of 10 CFR 50.160(b)(1)(i)–(iv)(A). To determine if the LMGS CP application has provided the minimum 10 CFR 50.160(b)(1) PSAR performance-based emergency preparedness design information, as required by 10 CFR 50.34(a)(10)(ii), the NRC staff reviewed the PSAR section 11.4 descriptions of the LMGS site preliminary plans for coping with emergencies, to determine if the minimum emergency preparedness design information was provided in the PSAR to issue a CP under 10 CFR 50.35(a).

The NRC staff evaluated the LMGS site performance-based emergency preparedness preliminary plans as follows:

1) Pursuant to 10 CFR 50.160(b)(1)(iii)(A)-(H), the NRC staff evaluated what the LMGS facility preliminary emergency response performance capabilities are for the LMGS emergency response team;

2) Once the LMGS emergency response function capabilities were determined, the NRC staff evaluated this information against the guidance of RG 1.242, to determine whether the required 10 CFR 50.34(a)(10)(ii) minimum PSAR preliminary plan information was included in the CP application.

The following SE subsections detail the NRC staff's review, evaluation, and regulatory findings of whether the requisite minimum PSAR preliminary plan information was provided in the LMGS CP application.

11.4.1.2.1 LMGS Emergency Response Function Evaluation (10 CFR 50.160(b)(1)(iii))

Pursuant to 10 CFR 50.160(b)(1)(iii), the emergency response team (i.e., ERO) must have sufficient capability to demonstrate the enumerated emergency response functions using drills or exercises. The emergency plan must include descriptions of how the ERO has been provided with sufficient capability to demonstrate the emergency response functions using drills or exercises.

The following SE sections document the NRC staff's review, evaluation, and regulatory findings of the requisite LMGS ERO minimum emergency preparedness preliminary plan design information included in the LMGS CP application PSAR.

Command and Control, ERO Staffing and Operations, and Critique and Corrective Actions Emergency Response Function Capabilities Evaluation (10 CFR 50.160(b)(1)(iii)(D); 10 CFR 50.160(b)(1)(iii)(E); 10 CFR 50.160(b)(1)(iii)(H))

ERO Defined Roles and Responsibilities

The NRC staff used the guidance of RG 1.242, section C.6.f, "Command and Control," to determine whether the LMGS emergency preparedness preliminary plan design descriptions for the requisite PSAR minimum design information are consistent with RG 1.242 guidance, and whether they adequately address the minimum PSAR information guidance of DANU-ISG-2022-01. RG 1.242, section C.6.f states that the emergency plan should describe the ERO capabilities to perform adequate command and control the supporting organizational structure with defined roles, responsibilities, and authorities for directing and performing emergency response functions, and RG 1.242, section C.6.g, "Staffing and Operations," states that the emergency plan should describe the capabilities to adequately staff the ERO emergency response functions within an appropriate timeframe and to include processes, equipment, facilities, and personnel. The emergency plan must also identify the processes used to complete training before assigning roles and responsibilities to the emergency response team.

The NRC staff's review determined that the normal facility operations staff, specifically the ERO qualified on-shift personnel, will initially assume the roles of the ERO personnel upon an emergency event declaration. PSAR section 11.4.3.2 states that during normal operations, the shift supervisor, who is the senior individual on shift in the main control room (MCR), will initially

assess, classify, and declare an emergency. Upon this official shift supervisor emergency declaration, other ERO qualified on-shift LMGS facility personnel will transition to their designated ERO responsibilities. LMGS facility preliminary ERO roles and responsibilities during a declared emergency are provided as follows:

- The shift supervisor will assume the role of ERO emergency director (ED) and is responsible for ERO performance and emergency response command and control to ensure that the emergency response functions are performed. The designated ED will remain in this role until relieved by the oncoming shift supervisor.

- The designated on-shift control room operator will assume the role of ERO operations coordinator (OC) and is responsible for providing oversight of the operation of the affected reactor unit(s) and providing operational assistance to the ED.

- A designated on-shift plant field technician (PFT) will assume the role of ERO radiation safety coordinator (RSC) and provide overall responsibility for radiological health physics aspects during the emergency.

- A separately designated on-shift PFT will assume the role of ERO field coordinator (FC) and is responsible for oversight of equipment status and returning equipment to service or replacing equipment damaged during the emergency.

- Additional designated ERO personnel, based on technical requirements of the position, may augment on-shift ERO personnel providing special expertise deemed beneficial, but not mandatory, to the emergency response, if required.

ERO Analysis to Determine the Minimum ERO Positions

PSAR section 11.4.3 states that the LMGS ERO composition, structure, and staffing (including defined positions, associated ERO roles and responsibilities, and timing of ERO augmentation), will be developed based on an analysis of the emergency response functions needed to respond to emergencies. The minimum ERO staff required to respond to abnormal and emergency operations is maintained at the LMGS site on a continuous basis. The LMGS ERO staff analysis will be provided in the LMGS emergency plan and FSAR that would be submitted to the NRC for review and evaluation as part of the submittal of a future LMGS OL application.

ERO Use of Emergency Plan Implementing Procedures (EPIPs)

The NRC staff used the guidance of RG 1.242, section C.6.f(3) to determine whether the LMGS emergency preparedness preliminary plan design descriptions are consistent with RG 1.242, and whether they adequately address the requisite PSAR minimum design information guidance of DANU-ISG-2022-01. RG 1.242, section C.6.f(3) states that the emergency plan should describe qualitative criteria to assess command and control, such as how the emergency response team leader would use EPIPs to demonstrate certain abilities such as the ability to: (1) lead and direct the team to perform emergency response functions, and (2) maintain awareness of the emergency conditions.

The NRC staff's review finds that PSAR section 11.4 provides adequate design descriptions of how EPIPs will be used to direct and facilitate command and control of ERO emergency response functions, consistent with RG 1.242 and as required for a CP application. The PSAR listed the following emergency response functions and tasks that will be directed or governed via EPIPs and the emergency plan during an emergency:

- Emergency classifications and emergency action levels (EALs) will be reflected in the appropriate EPIPs.
- EPIPs will be used to help facilitate the continual assessment, monitoring, and reporting to appropriate response personnel regarding the extent and magnitude of radiological conditions.
- Direction on the methods used to notify the NRC and the information to be provided to the NRC will be provided by EPIPs.
- EPIPS will direct the performance of protective measures within the site and EPZ boundary.
- Details related to site evacuation (including criteria for ordering an evacuation), assembly area locations, and monitoring of evacuees for contamination and control measures if contamination is found, will be described in EPIPs and the LMGS emergency plan.
- The arrangements for medical treatment and decontamination of onsite personnel who are injured or contaminated, and arrangements and agreements with ambulance services and primary and backup offsite medical facilities with adequately trained staff and facilities for handling potentially contaminated injured individuals.

ERO Emergency Preparedness and Response Training Process

The NRC staff used the guidance of RG 1.242, section C.6.g(2) to determine whether the LMGS emergency preparedness preliminary plan design descriptions are consistent with RG 1.242, and whether they address the requisite PSAR minimum design information guidance of DANU-ISG-2022-01. RG 1.242, section C.6.g(2) states that the emergency plan should describe, in part, the process used to complete training before assigning roles and responsibilities to the ERO.

PSAR section 11.4.7 states that emergency preparedness and response training is designed to ensure that LMGS ERO personnel are capable and qualified to perform their assigned duties under the emergency plan. This training includes general training on the LMGS emergency plan and EPIPs, as well as training specific to the individual's assigned emergency response functions. ERO personnel receive initial training before assuming assigned ERO responsibilities, and periodic continuing and re-qualification training thereafter. Initial and continuing training includes emergency preparedness drills and exercises.

PSAR section 11.4.7 also states that the methodology for developing and implementing the LMGS emergency preparedness and response training operational program utilizes an SAT as defined in 10 CFR 55.4, “Definitions.” PSAR section 11.3.1.1 states that training requirements are developed, implemented, and maintained using a systematic analysis of job and task requirements with a focus on the relationship between personnel activities and the associated human-system interfaces needed to accomplish those activities.

ERO Use of Drills and Exercises

The NRC staff used the guidance of RG 1.242, section C.6.g(2)(a) to determine whether the LMGS emergency preparedness preliminary plan design descriptions are consistent with RG 1.242, and whether they address the requisite PSAR minimum design information guidance of DANU-ISG-2022-01. RG 1.242, section C.6.g(2)(a) states that the emergency plan should describe the drills to be used to train and retrain facility personnel in emergency response responsibilities.

PSAR section 11.4.10 states that drills and exercises will be used to determine acceptable performance of emergency response functions, including the means for determining quality and timeliness to demonstrate the effectiveness of the emergency plan implementation. Drills and exercises, as well as responses to actual declared emergency events, are critiqued to identify improvement opportunities in ERO performance and the emergency plan implementation. Any weaknesses or deficiencies are evaluated, tracked, and corrected via the corrective action program.

PSAR section 11.4.7 states that the periodicity of continuing training drills and exercises will be determined by the results of the SAT process described in PSAR section 11.3.

ERO Capabilities to Critique Emergency Response Functions and Implement Corrective Actions

The NRC staff used the guidance of RG 1.242, section C.6.j(1), “Critique and Corrective Actions,” to determine whether the LMGS emergency preparedness preliminary plan design descriptions are consistent with RG 1.242, and whether they address the requisite PSAR minimum design information guidance of DANU-ISG-2022-01. RG 1.242, section C.6.j(1) states that the emergency plan should describe the capabilities to critique emergency response functions and implement effective corrective actions.

PSAR section 11.4.10 states that drills and exercises, as well as responses to actual declared emergency events, are critiqued to identify improvement opportunities in ERO performance and the emergency plan implementation, and that any weaknesses or deficiencies will be evaluated, tracked, and corrected via the corrective action program.

Staff Findings and Conclusions for the Command and Control, ERO Staffing and Operations, and Critique and Corrective Actions Emergency Response Function Capabilities Evaluation

Pursuant to 10 CFR 50.34(a)(10)(ii), each PSAR for a CP application involving a small modular reactor, non-LWR, or non-power production or utilization facility requires a discussion of an applicant’s preliminary plans for coping with emergencies. Based on the information provided above, the NRC staff identified that the LMGS CP application PSAR is consistent with RG 1.242 and contains the minimum preliminary plan information because the PSAR includes the

proposed command and control, ERO staffing and operations, and critique and corrective actions emergency response functional capabilities. The NRC staff also confirmed that the PSAR addresses the minimum information provided in DANU-ISG-2022-01 and is therefore acceptable. Thus, the NRC staff concludes that the above LMGS PSAR preliminary command and control, ERO staffing and operations, and critique and corrective actions emergency response functions capability descriptions meet the emergency preparedness preliminary plan requirement of 10 CFR 50.34(a)(10)(ii).

The final LMGS site emergency plan ERO command and control, staffing operations, and critique and corrective actions emergency response functions capability design details necessary for the NRC staff to evaluate compliance with the applicable emergency response function capabilities criteria in 10 CFR 50.160(b)(1)(iii), would be provided by the applicant through the submittal of the final LMGS site emergency plan and FSAR as part of the LMGS OL application, should the applicant apply for an OL. At that time, the NRC staff would perform the requisite detailed evaluation of the LMGS ERO command and control, staffing and operations, and critique and corrective actions final emergency response function design descriptions under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

ERO Communications, Communications with the NRC, and Radiological Assessment Emergency Response Functions, and Emergency Response Facility Design, Capability Evaluation (10 CFR 50.160(b)(1)(iii)(C), 10 CFR 50.160(b)(1)(iii)(F), 10 CFR 50.160(b)(1)(iv)(A)(3), 10 CFR 50.160(b)(1)(iv)(A)(4))

ERO Emergency Response Facility Design

The NRC staff used the guidance of RG 1.242, section C.7.d to determine whether the LMGS emergency preparedness preliminary plan design descriptions are consistent with RG 1.242, and whether they address the requisite PSAR minimum design information guidance of DANU-ISG-2022-01. RG 1.242, section C.7.d states that the emergency plan should describe each emergency response facility (ERF) including, as applicable, the descriptions of ERF location, capabilities, size, equipment, and backup locations to transfer the functions if the facility is not habitable or accessible. The NRC staff's review and evaluation for 10 CFR 50.160(b)(1)(iii)(F)(2), "Protective equipment," is documented in section titled "Protective Action Emergency Response Function and Protective Equipment Usage Capabilities Evaluation (10 CFR 50.160(b)(1)(iii)(B); 10 CFR 50.160(b)(1)(iii)(F)(2))," of this SE.

PSAR section 11.4.8.1 states that the LMGS facility design includes both primary and backup ERFs from which emergency response command and control can be exercised during an emergency. The LMGS facility MCR will serve as the primary ERF and will be located in the controls and electrical building (CEB), and the backup ERF will be located in the engineering, administration, and training building (EATB). The CEB and the EATB are shown in PSAR figure 1.1-4. PSAR section 11.4.8.1 also states that the primary and backup ERFs are designed and equipped with functionally similar systems and equipment needed for the ERO to provide direction and control during an emergency. In the event of an emergency declaration, the MCR becomes the primary ERF and will be the communications center for the LMGS emergency response. The backup ERF will be used if the primary ERF is uninhabitable or inaccessible.

PSAR section 11.4.8.1 further states that the ERFs will be designed with:

- Reliable voice and data communication systems and equipment for ERO communications as well as timely notifications to response personnel and offsite agencies (see this SE subsection titled "*ERO Communications Design Capabilities*" for more details).

- Displays, alarms, and indications needed to assess and monitor radiological and plant conditions during an emergency event (see this SE subsection titled "*Radiological Conditions and Releases, and Core or Vessel Damage Assessment Capabilities*" for more details).

ERO Communications Design Capabilities

The NRC staff used the guidance of RG 1.242, section C.6.e, "Communications," and section 7.c, "Communications with the NRC," to determine whether the LMGS emergency preparedness preliminary plan design descriptions are consistent with RG 1.242, and whether they address the requisite PSAR minimum design information guidance of DANU-ISG-2022-01. RG 1.242, section C.6.e states that the emergency plan should describe the capabilities to establish and maintain communications among the response facilities and with organizations that may have emergency response responsibilities. RG 1.242, section 7.c states that the emergency plan should describe the planning activities, capabilities, and processes or procedures to support emergency notifications and communication and data transmission utilizing technology systems.

PSAR section 11.4.8.3 states that the voice and data equipment that will provide the communications capability to notify and activate the ERO regarding the status of an emergency to facilitate performance of emergency response protective actions, is provided by both the primary and backup ERF. The ERFs will be equipped with communications capabilities to collect, process, display, share, and communicate LMGS plant parameters to aid in efficient emergency response decision-making. The ERFs voice and data communications capabilities will include backup communication equipment to ensure continuous communications capability.

Additional LMGS site ERO communications capabilities will provide:

- Timely notifications and real-time around-the-clock plant response data sharing with the NRC, as well as communications with other offsite agencies, stakeholders, and support organizations.

- Coordination with the NRC and state and local offsite agencies to ensure that LMGS site voice and data communications systems and equipment are compatible with those of the offsite agencies.

- Periodic communication checks with offsite agencies, as appropriate, to verify communication system and equipment operability.

The final LMGS site ERO communication systems and equipment design details would be described in the LMGS site emergency plan that will be submitted as part of the LMGS OL application, should the applicant apply for an OL.

Radiological Conditions and Releases, and Core or Vessel Damage Assessment Capabilities

The NRC staff used the following guidance to determine whether the LMGS emergency preparedness preliminary plan design descriptions are consistent with RG 1.242, and whether they address the requisite PSAR minimum design information guidance of DANU-ISG-2022-01:

- RG 1.242, section C.6.h(1), "Radiological Conditions," provides that the emergency plan should describe the capabilities to assess, monitor, and report to applicable response personnel the radiological conditions in and around the facility and onsite locations.

- RG 1.242, section C.6.h(3)(a), "Core or Vessel Damage," provides that the emergency plan should describe the capabilities to assess, monitor, and report to the applicable response personnel the extent of any core damage.

- RG 1.242, section C.6.h(4)(a), "Releases," provides that the emergency plan should describe the capabilities to assess, monitor, and report to the applicable response personnel the extent of any radiological releases including the releases of hazardous chemicals produced from licensed material.

The capability to report to ERO personnel (including the radiological conditions in and around the LMGS facility), is reviewed in the "ERO Communications Design Capabilities" subsection of this SE section. PSAR section 11.4.8.2 describes the LMGS radiological condition indications that will be provided to the ERO t to assess and monitor radiological conditions in and around the LMGS facility during emergencies. The facility procedures and installed equipment facilitate the continual assessment, monitoring, and reporting, to the ERO, of the extent and magnitude of radiological conditions, including:

- Radiological releases.
- Releases of hazardous chemicals produced from licensed material.
- Damage to the reactor core or reactor vessel.
- Damage to other nuclear fuel-containing vessels.

PSAR section 11.4.8.1 states that to perform the required emergency response functions, displays, alarms, and indications to assess and monitor radiological and plant conditions during an emergency event, these emergency planning activities will be provided in the LMGS ERFs for the ERO to provide effective direction during an emergency.

Details on the LMGS radiological assessment equipment, including equipment locations and alarm and indication capabilities, would be described in the final LMGS emergency plan to be provided as part of the LMGS OL application, should the applicant apply for an OL.

NRC Staff Findings and Conclusions for the ERO Communications, Communications with NRC, and Radiological Assessment Emergency Response Functions, and ERF Design Capability Evaluation

Pursuant to 10 CFR 50.34(a)(10)(ii), an applicant's PSAR for a CP application shall include a discussion of the preliminary plans for coping with emergencies. Based on the NRC staff's review above, the NRC staff determined that, in accordance with RG 1.242, the LMGS CP application PSAR is consistent with and contains the minimum preliminary plan information

because the PSAR included the proposed communications, radiological assessment emergency response function capabilities, and ERFs design and capability descriptions for a CP application as documented above. The NRC staff also confirmed that the LMGS PSAR addresses the minimum information provided in DANU-ISG-2022-01 for a CP application and is therefore acceptable. Thus, the NRC staff concludes that the LMGS PSAR preliminary communications, radiological assessment emergency response functions, and ERFs design and capability descriptions meet the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii).

The final LMGS emergency plan radiological assessment and communications emergency response function capability, as well as the ERFs capabilities and design details, necessary for the NRC staff to perform an SE for compliance with the applicable emergency response function capabilities criteria in 10 CFR 50.160(b)(1)(iii) and 10 CFR 50.160(b)(1)(iv)(A), would be provided by the applicant through the submittal of the final LMGS emergency plan and FSAR as part of the LMGS OL application, should the applicant apply for an OL. At that time, NRC staff would perform a more detailed evaluation of the radiological assessment, communications emergency response functions and ERF final design descriptions under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

Event Classification and Mitigation and Reentry Emergency Response Functions Capabilities Evaluation (10 CFR 50.160(b)(1)(iii)(A), 10 CFR 50.160(b)(1)(iii)(G))

Capability to Assess and Monitor

The NRC staff used the guidance of RG 1.242, section C.6.c(1), to determine whether the LMGS emergency preparedness preliminary plan design descriptions are consistent with RG 1.242, and whether they address the requisite PSAR minimum design information guidance of DANU-ISG-2022-01. RG 1.242, section C.6.c(1) states that the emergency plan should describe the capabilities to perform event classification and mitigation including the methods, processes, equipment, specific instruments, parameters, facilities, and personnel.

PSAR section 11.4.8.1 states that to perform the required emergency response functions, the LMGS ERFs for the ERO will include displays, alarms, and indications to assess and monitor radiological and plant conditions during an emergency event to provide effective direction during an emergency. The LMGS ERFs will be provided with capabilities to collect, process, display, share, and communicate plant parameters. Further, PSAR section 2.3.3, “On-site Meteorological Measurements Program,” states that the applicant plans to construct and install an onsite meteorological tower which will be the primary source of meteorological data for operations at the LMGS facility. Therefore, the NRC staff finds that the capabilities for the LMGS ERO personnel to assess, monitor, and perform emergency event classification and mitigation, will be provided in the LMGS ERFs.

Capability to Classify

The NRC staff used the guidance of RG 1.242, section 6.c(3), to determine whether the LMGS emergency preparedness preliminary plan design descriptions are consistent with RG 1.242, and whether they address the requisite PSAR minimum design information guidance of DANU-ISG-2022-01. RG 1.242, section 6.c(3) states that the emergency plan should describe the emergency classification scheme, the associated EALs applicable for the design of the facility,

and the technical basis and methodology for determining the thresholds corresponding to each of the EALs.

PSAR section 11.4.5.2 states that the technical basis and methodology for developing an LMGS facility design and site-specific emergency classification scheme and associated EALs will be submitted as part of the LMGS OL application, should the applicant apply for an OL, and will be reflected in the appropriate EPIPs. The NRC staff determined that this information is acceptable because the final LMGS facility site-specific design, that would be provided in an FSAR, would be necessary to develop the final emergency event classification action levels, and this final site-specific, FSAR, design information, can reasonably be left for later consideration as stated in the CP issuance requirement in 10 CFR 50.35(a)(2). Further, PSAR section 11.4.5.2 states that the resultant LMGS emergency classifications and EALs will be provided in the final LMGS emergency plan to be submitted as part of any OL application.

Reentry and Capability to Repair LMGS Facility Malfunctions to Return Facility to Safe Conditions

The NRC staff used the guidance of RG 1.242, section C.6.i and section C.6.c(2), to determine whether the LMGS emergency preparedness preliminary plan design descriptions are consistent with RG 1.242, and whether they address the requisite PSAR minimum design information guidance of DANU-ISG-2022-01. RG 1.242, section C.6.i states that the emergency plan should describe the capabilities to develop and implement reentry plans for access to the facility after radiological emergencies as well as the capabilities to develop and implement reentry plans for access to the facility following hostile-action-based emergencies. Further, RG 1.242, section C.6.c(2), states, in part, that the emergency plan should describe how the emergency response team will repair facility malfunctions in a timely manner and return the facility to safe conditions, which includes termination of accident progression and return of the facility to within safety limits or technical specifications.

PSAR section 11.4.3.2 states that the ED is authorized to declare the LMGS safe for reentry. Specifically, the RSC and the security manager will advise the ED during radiological and hostile-action emergencies when the LMGS is safe for reentry. Likewise, to repair the LMGS facility to safe conditions, the OC and the FC will advise the ED on the status of LMGS plant SSCs during an emergency. If required, the ED will make the request for additional personnel to repair the facility to safe conditions. The detailed plans and procedures for reentry will be provided in the LMGS OL application submittal, should the applicant apply for an OL.

NRC Staff Findings and Conclusions for the Event Classification and Mitigation and Reentry Emergency Response Functions Capabilities Evaluation

Pursuant to 10 CFR 50.34(a)(10)(ii), each PSAR for a CP application requires a discussion of the preliminary plans for coping with emergencies. Based on the information provided above, the NRC staff determined that the LMGS CP application PSAR is consistent with and contains the minimum RG 1.242 PSAR preliminary plan information because the PSAR included the proposed event classification and mitigation and reentry emergency response function capabilities and addressed the minimum PSAR information guidance of DANU-ISG-2022-01 and is therefore acceptable. Thus, the NRC staff concludes that the above LMGS PSAR preliminary event classification and mitigation and reentry capability descriptions meet the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii).

The final LMGS emergency plan event classification and mitigation and reentry emergency response function capability design details, necessary to demonstrate compliance with the applicable emergency response function capabilities criteria in 10 CFR 50.160(b)(1)(iii), would be provided by the applicant through submittal of the final LMGS emergency plan and FSAR as part of an OL application. At that time, the NRC staff would perform a more detailed evaluation of the event classification and mitigation and reentry emergency response functions final design descriptions under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

Protective Action Emergency Response Function and Protective Equipment Usage Capabilities Evaluation (10 CFR 50.160(b)(1)(iii)(B); 10 CFR 50.160(b)(1)(iii)(F)(2))

The NRC staff used the guidance of RG 1.242, section C.6.d, “Protective Actions,” and section C.6.h(2)(a), “Protective Equipment,” to determine whether the LMGS emergency preparedness preliminary plan design descriptions are consistent with RG 1.242, and whether they address the requisite PSAR minimum design information guidance of DANU-ISG-2022-01. RG 1.242, section C.6.d, states that the emergency plan should describe the capabilities to determine, implement, and recommend appropriate protective actions for a variety of hazards, that include the methods, processes, equipment, facilities, and personnel, to implement protective action emergency response function capabilities. RG 1.242, section C.6.h(2)(a), states that the emergency plan should describe the capabilities to issue and use protective equipment and the capabilities to expand mitigation and protective action strategies that include the methods, processes, equipment, facilities, and personnel.

PSAR section 11.4.5.1 states that the performance of protective actions, coordination of first aid administration, and emergency transport of personnel, will be governed by EPIPs and that protective actions will be communicated via onsite communication systems with notifications to impacted onsite personnel. PSAR section 11.4.6 states that agreements are established with offsite medical facilities and ambulance services for onsite personnel who are injured or contaminated, and for personnel who may have received significant radiation exposures of radioactive material. Further, the PSAR listed onsite LMGS facility protective action and offsite emergency preparedness mitigation capabilities include:

- Evacuating or sheltering in-place of plant personnel if conditions warrant (PSAR section 11.4.5.1).
- Specifying proper personal protective equipment staged at designated onsite locations (PSAR section 11.4.5.1).
- Provisions to evaluate radiological contamination prior to transport to a medical facility (PSAR section 11.4.6).
- Establishing onsite first aid capability with adequate medical equipment and supplies appropriately located throughout the plant (PSAR section 11.4.6).
- The capability to coordinate offsite medical treatment and emergency transport of injured or contaminated personnel offsite for medical care (PSAR section 11.4.5.1).
- Establishing contamination control measures and onsite decontamination facilities and equipment (PSAR section 11.4.5.1).

- The ERO will have the capability to assess plant radiological conditions to support recommending protective actions to offsite authorities as conditions warrant (PSAR section 11.4.5).

LMGS emergency plan EPIPs will govern the performance of protective measures within the site and the PEP EPZ boundary and the coordination of first aid administration.

PSAR section 11.4.8.2 states that emergency response equipment will also include portable and fixed radiation monitors and sampling equipment located throughout the plant and that the fixed radiation monitors will provide indications in the ERFs.

The details on this protective equipment, including equipment locations and alarm and indication capabilities, will be described in the final LMGS emergency plan that will be submitted as part of the LMGS OL application. The final LMGS emergency plan will also describe the agreements with state and local governmental agencies, as well as other local organizations (e.g., law enforcement, medical, hospital, ambulatory transport services, fire response) for notification and emergency response support that may provide LMGS emergency plan implementation support. PSAR section 11.4.5.1 states that the details related to ordering a site evacuation, assembly area locations, and monitoring of evacuees for contamination and the appropriate control measures, if contamination is found, will be described in the final LMGS emergency plan to be submitted with the LMGS OL application.

NRC Staff Findings and Conclusions for the Protective Action Emergency Response Function and Protective Equipment Usage Capabilities Evaluation (10 CFR 50.160(b)(1)(iii)(B); 10 CFR 50.160(b)(1)(iii)(F)(2))

Pursuant to 10 CFR 50.34(a)(10)(ii) an applicant's PSAR for a CP application must include a discussion of the preliminary plans for coping with emergencies. Based on the information provided above, the NRC staff determined that the LMGS CP application PSAR is consistent with and contains the minimum RG 1.242 PSAR preliminary plan information because the PSAR included proposed protective action emergency response functions and protective equipment usage capabilities. The NRC staff also determined that the PSAR addresses the minimum information guidance of DANU-ISG-2022-01 and is therefore acceptable. Thus, the NRC staff concludes that the above LMGS PSAR preliminary protective action emergency response function and protective equipment usage capability descriptions meet the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii).

The final LMGS emergency plan protective action emergency response function and protective equipment usage capabilities design details, necessary to demonstrate compliance with the applicable emergency response function capabilities criteria in 10 CFR 50.160(b)(1)(iii) would be provided by the applicant through the submittal of the final LMGS emergency plan and FSAR as part of an OL application. At that time, the NRC staff would perform a more detailed evaluation of the protective action emergency response function and protective equipment usage capabilities final design descriptions under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

NRC Staff Findings and Evaluation Conclusions of the LMGS ERO Emergency Response Performance Evaluation (10 CFR 50.160(b)(1)(iii))

Based on the NRC staff's review and findings of the PSAR emergency preparedness preliminary plan design discussion and descriptions of the LMGS ERO emergency response functions capabilities, the ERF design, and the protective equipment usage capabilities, the NRC staff determined, because this information was included in the PSAR sections discussed, that the PSAR preliminary plan descriptions of the LMGS ERO emergency response functions and protective equipment usage capabilities which are required in 10 CFR 50.160(b)(1)(iii)(A)-(H), and the communications with NRC required planning activities in 10 CFR 50.160(b)(1)(iv)(A)(3), and the required ERF design planning activities in 10 CFR 50.160(b)(1)(iv)(A)(4)), address the necessary minimum information described in DANU-ISG-2022-01 and is therefore acceptable. Thus, the NRC staff concludes that the above LMGS PSAR ERO emergency response functions capability and ERF design descriptions meet the emergency preparedness preliminary plan discussion requirement in 10 CFR 50.34(a)(10)(ii).

11.4.1.2.2 LMGS Emergency Response Performance Objectives Evaluation (10 CFR 50.160(b)(1)(ii))

Pursuant to 10 CFR 50.160(b)(1)(ii) an applicant or licensee's emergency plan must describe a performance monitoring program to develop and maintain at the beginning of each calendar quarter, a list of performance objectives for that calendar quarter and maintain records showing the implemented performance objectives and associated metrics during each calendar quarter for the previous eight calendar quarters. In addition, the applicant or licensee should describe the implementation of a performance objective scheme that reflects the enumerated emergency response functions listed in 10 CFR 50.160(b)(1)(iii)(A)-(H) and the performance measures used during drills and exercises to determine acceptable ERO performance, including the means for determining quality and timeliness.

The NRC staff used RG 1.242, section C.6.b, "Performance Objectives," to determine whether the LMGS emergency preparedness preliminary plan design descriptions are consistent with RG 1.242 and address the requisite PSAR minimum design information guidance of DANU-ISG-2022-01. RG 1.242, section C.6.b provides that guidance for the performance objective topics that should be included in the emergency plan's description of the licensee's performance monitoring program.

PSAR section 11.4.10 states that to demonstrate the effectiveness of the emergency plan implementation, the LMGS performance monitoring program will provide quarterly updates of the performance objectives and metric data, and maintenance of these objectives and data for the previous eight calendar quarters. The LMGS emergency plan will describe the emergency response performance monitoring program design elements such as:

The process used to develop performance metrics and objectives for each LMGS emergency response function in 10 CFR 50.160(1)(iii), including the methodology used to develop the performance objectives, the basis for relying on the objectives, and descriptions of how performance metrics and objectives acceptability or successful achievement is determined.

- Descriptions of how the use of drills and exercises determine acceptable performance of emergency response functions including the means for determining quality and timeliness.

- Benchmark reference levels to evaluate performance of 10 CFR 50.160(b)(1)(iii) emergency response functions.

- Performance objective data format and correction of previous performance objective data.

Drills and exercises, as well as responses to actual declared emergency events, are critiqued to identify improvement opportunities in ERO performance and the LMGS emergency plan implementation. Any weaknesses or deficiencies are evaluated, tracked, and corrected via the corrective action program.

NRC Staff Findings and Evaluation Conclusions of the LMGS Emergency Response Performance Objectives Evaluation (10 CFR 50.160(b)(1)(ii))

Pursuant to 10 CFR 50.34(a)(10)(ii) an applicant's PSAR must include a discussion of the preliminary plans for coping with emergencies. Based on the information provided above, the NRC staff determined that the LMGS PSAR is consistent with and contains the minimum RG 1.242 preliminary plan information because the PSAR included the proposed LMGS performance monitoring program descriptions and addresses the minimum PSAR information guidance of DANU-ISG-2022-01 and is therefore acceptable. Thus, the NRC staff concludes that the LMGS PSAR preliminary performance monitoring program design descriptions meet the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii) and is sufficient for the issuance of a CP under 10 CFR50.35, as applicable.

The final LMGS facility emergency plan performance objectives and performance monitoring program design details, necessary to demonstrate compliance with the applicable performance objectives criteria in 10 CFR 50.160(b)(1)(ii), would be provided by the applicant through the submittal of the final LMGS emergency plan and FSAR as part of an OL application, where the NRC staff will perform a more detailed evaluation of the final performance objectives design details under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

11.4.1.2.3 Maintenance of Performance and Methods for Maintaining the Emergency Plan Evaluation (10 CFR 50.160(b)(1)(i), 10 CFR 50.160(b)(1)(iv)(A)(6))

The NRC staff used the guidance of RG 1.242, section C.6.a(2), "Maintenance of Performance," and RG 1.242, section 7.f, "Emergency Plan Maintenance," to determine whether the LMGS emergency preparedness preliminary plan design descriptions are consistent with RG 1.242 and address the requisite PSAR minimum design information criteria in 10 CFR 50.34(a)(10) and the guidance of DANU-ISG-2022-01.

PSAR section 11.4.3.1 states that the LMGS site OC will be responsible for the maintenance of the emergency plan. The OC is shown in figure 11.1-2, "Construction Organization." PSAR section 11.4.10 states that the maintenance of emergency plan performance objectives will be demonstrated through completion of the performance objectives for each LMGS emergency response function in 10 CFR 50.160(b)(1)(iii).

The emergency response procedures (e.g., EPIPs) will be maintained and controlled onsite by the OC. Following procedure direction in EPIPs, changes to emergency response procedures and LMGS facility changes, will be reviewed to ensure there is not a reduction in emergency plan effectiveness in accordance with 10 CFR 50.54(q), "Emergency plans." Periodic independent reviews will be performed for the LMGS emergency plan. Further, pursuant to 10 CFR 50.34(b)(6)(v)(B), the details of the frequency of the emergency plan independent reviews would be provided in the final LMGS emergency plan that will be submitted in an OL application.

NRC Staff Findings and Evaluation Conclusions of the LMGS Maintenance of Performance and Methods for Maintaining the Emergency Plan Evaluation (10 CFR 50.160(b)(1)(i), 10 CFR 50.160(b)(1)(iv)(A)(6))

Pursuant to 10 CFR 50.34(a)(10) an applicant's PSAR must include a discussion of an applicant's preliminary plans for coping with emergencies. Based on the NRC staff's review and documentation of requisite PSAR minimum information provided above, the NRC staff determined that the LMGS PSAR is consistent with and contains the minimum RG 1.242 preliminary plan information because the PSAR included the discussions of the proposed maintenance of performance and methods for maintaining the emergency plan and addresses the minimum PSAR information guidance of DANU-ISG-2022-01 and is therefore acceptable. Thus, the NRC staff concludes that the above LMGS PSAR preliminary maintenance of performance and emergency plan maintenance methods meet the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii).

The final LMGS emergency plan maintenance of performance and emergency plan evaluation methods design details, necessary to demonstrate compliance with the applicable maintenance of performance criteria in 10 CFR 50.160(b)(1)(i) and the emergency plan maintenance planning activities criteria in 10 CFR 50.160(b)(1)(iv)(A), would be provided by the applicant through the submittal of the final LMGS emergency plan and FSAR as part of an OL application. At that time, the NRC staff would perform a more detailed evaluation of the maintenance of performance and emergency plan maintenance final design details under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

11.4.1.2.4 Onsite Planning Activities Evaluation (10 CFR 50.160(b)(1)(iv)(A))

The emergency preparedness related onsite planning activities enumerated in 10 CFR 50.160(b)(1)(iv)(A) are activities that are not readily observable or effectively measured through drills and exercises. The NRC staff used the onsite planning activities guidance of RG 1.242, section C.7, "Onsite Planning Activities," to determine whether the LMGS emergency preparedness preliminary plan design descriptions are consistent with RG 1.242 and address the requisite PSAR minimum design information guidance of DANU-ISG-2022-01.

Further, the NRC staff evaluation of the "Communications with the NRC" planning activity (10 CFR 50.160(b)(1)(iv)(A)(3)) and the ERF design planning activity (10 CFR 50.160(b)(1)(iv)(A)(4)) was performed in SE section 11.4.1.2.1, and the "Emergency Plan Maintenance" planning activity (10 CFR 50.160(b)(1)(iv)(A)(6)) evaluation was performed by the NRC staff in SE section 11.4.1.2.3.

Therefore, the following SE subsections document the LMGS PSAR discussion of onsite planning activities emergency preparedness preliminary plan design information that addresses the applicable planning activities guidance.

Prepare and Issue Public Information During Emergencies (10 CFR 50.160(b)(1)(iv)(A)(1))

PSAR section 11.4.4.1 states that the LMGS emergency plan will include requirements for making appropriate notifications to the NRC at the onset of and during an emergency event. The section also identifies additional federal resources that are available to provide ingestion pathway emergency event response support. PSAR section 11.4.4.2 states that the appropriate notification requirements, communication and procedural protocols and agreements will be developed and established in coordination with State and local agencies, public safety officials, and support organizations within the immediate vicinity of the LMGS site. The section further states that the LMGS emergency plan will describe agreements or plans established with state and local governmental agencies for notification or response support in the event of an emergency or other local organizations that may provide support (e.g., law enforcement; medical, hospital, ambulance transport services; and fire response services).

Implement the NRC-approved Emergency Response Plan in Conjunction with the Licensee's Safeguards Contingency Plan (10 CFR 50.160(b)(1)(iv)(A)(2))

PSAR section 11.4.9 states that the LMGS emergency plan will be implemented in conjunction with the LMGS safeguards contingency plan.

Provide Site Familiarization Training for any Offsite Organization that may Respond to the Site in the Event of an Emergency (10 CFR 50.160(b)(1)(iv)(A)(5))

PSAR section 11.4.7 states that offsite agencies and individuals that may respond to the LMGS site in the event of an emergency will be provided with site familiarization training.

NRC Staff Findings and Evaluation Conclusions of the LMGS Onsite Planning Activities Evaluation (10 CFR 50.160(b)(1)(iv)(A))

Pursuant to 10 CFR 50.34(a)(10)(ii) an applicant's PSAR must include a discussion of the preliminary plans for coping with emergencies. Based on the information provided above, the NRC staff determined that the PSAR is acceptable because it includes discussions of the proposed onsite planning activities and addresses the minimum information necessary for a PSAR consistent with DANU-ISG-2022-01 and RG 1.242. Therefore, the NRC staff concludes that the above LMGS PSAR preliminary onsite planning activities descriptions meet the emergency preparedness preliminary plan discussion requirement in 10 CFR 50.34(a)(10)(ii).

The final LMGS facility emergency plan onsite planning activities design details, necessary for to demonstrate compliance with the applicable onsite planning activities criteria in 10 CFR 50.160(b)(1)(iv)(A), would be provided by the applicant through the submittal of the final LMGS emergency plan and FSAR as part an OL application, where NRC staff will perform a more detailed evaluation of the LMGS onsite planning activities final design descriptions under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

11.4.1.2.5 NRC Staff Findings and Evaluation Conclusions of the LMGS Emergency Preparedness Performance-based Framework (10 CFR 50.160(b)(1))

Based on the NRC staff's evaluation and findings of the LMGS PSAR preliminary plan principal design descriptions for (1) maintenance of emergency response performance and preparedness, (2) emergency response performance objectives and associated metrics development, (3) the LMGS ERO's emergency response function capabilities, and (4) the licensee's onsite emergency response planning activities design and implementation, the NRC staff concludes that, because the minimum information was included in the PSAR, the LMGS CP application's preliminary emergency response performance-based framework design descriptions meet the preliminary discussion requirement of 10 CFR 50.34(a)(10)(ii).

11.4.1.3 LMGS Site Hazard Analysis and Significant Impediments to the Development of the LMGS Emergency Plans Evaluation (10 CFR 50.160(b)(2); 10 CFR 100.21(g))

Pursuant to 10 CFR 50.160(b)(2) applicants are required to perform an analysis to assess any credible hazards that would adversely impact the implementation of emergency plans, of any facility contiguous to or near the applicant's proposed facility site. Pursuant to 10 CFR 100.21(g), applications for a site approval for a commercial power reactor shall identify the physical characteristics unique to the proposed site that could pose a significant impediment to the development of emergency plans.

The NRC staff used the guidance of RG 1.242, section C.9, "Hazard Analysis of Contiguous or Nearby Facilities," to review and evaluate the LMGS facility emergency preparedness preliminary plan design descriptions for requisite PSAR minimum design information. RG 1.242, section C.9 states that the emergency plan will describe the results of a hazard analysis of any contiguous or nearby facility in which a credible hazard has been identified that would adversely impact the implementation of the emergency plans. The section also provides that the emergency plan should describe planning activities or emergency response functions that will address any credible hazard that would adversely impact the implementation of emergency plans.

In addition, the NRC staff used the guidance of Standard Review Plan (SRP), Revision 3 (ML063410307) section 13.3, "Emergency Planning," section III, "Review Procedures," subsection "General Review Procedures," guidance item 5, to determine whether the LMGS emergency preparedness preliminary plan design descriptions are consistent with RG 1.242 and address the requisite PSAR minimum design information guidance of DANU-ISG-2022-01.The section states that the NRC staff should confirm that the application addresses the physical characteristics unique to the proposed site that could pose a significant impediment to the development of emergency plans, as required by 10 CFR 100.21(g). The section also provides that if such physical characteristics are identified, the NRC staff should determine whether the application identifies measures that would, when implemented, mitigate or eliminate the significant impediment.

PSAR section 2.2 states that facilities and transportation routes were identified and evaluated to determine the effects of potential accidents in the vicinity of the LMGS site and whether hazards due to the facilities and transportation routes need to be considered design-basis events for the LMGS plant design. Several identified major facilities and transportation routes are shown in

PSAR figure 2.2-1 and PSAR figure 2.2-2. The accident categories evaluated were explosions, flammable vapor clouds, fires, and toxic chemicals. Toxic and explosive chemicals produced or stored at the adjacent TDCC Seadrift Operations are included in PSAR table 2.2-2. Toxic and explosive chemicals transported via truck, rail, barge, and pipeline that were evaluated are included starting at PSAR table 2.2-3 through table 2.2-7. PSAR section 11.4.1 states that the hazard analysis required by 10 CFR 50.160(b)(2) will be conducted for the LMGS facilities to identify any credible hazard that would warrant specific emergency plan implementation measures and that the results of this analysis and resultant emergency plan implementation measures, will be addressed in the LMGS emergency plan that will be provided for evaluation as part of an OL application submittal.

PSAR section 11.4.1 states that the LMGS site has been evaluated for physical characteristics that could pose a significant impediment to the development of the LMGS emergency plans in accordance with 10 CFR 100.21(g). PSAR section 11.4.1 contains the results of the LMGS site physical characteristics evaluation are listed in and the conclusion that there are no physical characteristics unique to the LMGS site that could pose a significant impediment to the development of LMGS emergency plans in accordance with 10 CFR 100.21(g).

11.4.1.3.1 NRC Staff Findings and Evaluation Conclusions of the LMGS Site Hazard Analysis Results and Significant Impediments to the Development of the LMGS Emergency Plan Evaluation (10 CFR 50.160(b)(2); 10 CFR 100.21(g))

Pursuant to 10 CFR 50.34(a)(10)(ii) an applicant's PSAR must contain a discussion of the preliminary plans for coping with emergencies. Based on the information and documentation of requisite PSAR minimum information and evaluation provided above, the NRC staff determined that the LMGS PSAR is acceptable because it contains the minimum RG 1.242 preliminary hazard analysis results descriptions and the identification of physical characteristics unique to the LMGS proposed site that could pose significant impediments to the development of the LMGS emergency plan. The NRC staff also determined that the PSAR addressed the minimum PSAR information guidance of DANU-ISG-2022-01 and is therefore acceptable. Thus, NRC staff concludes that the above LMGS PSAR preliminary hazard analysis and emergency plan development impediments identification descriptions meet the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii).

Further, PSAR section 11.4.1 states that the results of the LMGS site physical characteristics evaluation concluded that there are no physical characteristics unique to the LMGS site that could pose a significant impediment to the development of LMGS site emergency plan. Therefore, the NRC staff determined that 10 CFR 100.21(g) is met.

The final LMGS emergency plan hazard analysis and confirmation of identification of impediments to emergency plan development final design details necessary to demonstrate compliance with the applicable regulatory criteria of 10 CFR 50.160(b)(2) and 10 CFR 100.21(g), would be provided by the applicant through the submittal of the final LMGS emergency plan and FSAR as part of an OL application, where the NRC staff would perform a more detailed evaluation of the LMGS facility site hazards and identification of unique site physical characteristics that may pose impediments to emergency plan development, final analysis results, under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

11.4.1.4 LMGS Ingestion Response Planning Evaluation (10 CFR 50.160(b)(4))

Pursuant to 10 CFR 50.160(b)(4), an applicant's emergency plan requires a description or reference of the ingestion response planning capabilities that provide actions to prevent contaminated food and water from entering into the ingestion pathway. The NRC staff used the guidance of RG 1.242, section C, guidance item 3, to determine whether the LMGS emergency preparedness preliminary plan design descriptions are consistent with RG 1.242 and address the requisite PSAR minimum design information guidance of DANU-ISG-2022-01. RG 1.242, section C.3 states that even if the facility's PEP EPZ is bounded by the site boundary, 10 CFR 50.160(b)(4) requires that the applicant and licensee must reference the capabilities of Federal, State, and local authorities that provide actions to prevent contaminated food and water from entering the ingestion pathway. The section also states that the capabilities described in the emergency plan need to address major exposure pathways associated with the ingestion of contaminated food and water.

PSAR section 11.4.2.2 states that, although the PEP EPZ coincides with the LMGS site boundary, the development of the LMGS emergency plan will include ingestion response planning as required by 10 CFR 50.160(b)(4). Specifically, the emergency plan will describe the capabilities that support actions to prevent contaminated food and water from entering the ingestion pathway in the event of a radiological emergency. These capabilities include those of applicable Federal, State, and local authorities to assess, sample, and notify to interdict food and water in a timely manner to avoid exceeding ingestion protective action guideline doses. PSAR section 11.4.4 identify several government agencies that may be requested to support the LMGS emergency plan ingestion pathway response.

11.4.1.4.1 NRC Staff Findings and Evaluation Conclusions of the LMGS Ingestion Response Planning Evaluation (10 CFR 50.160(b)(4))

Pursuant to 10 CFR 50.34(a)(10)(ii) an applicant's PSAR requires a discussion of the preliminary plans for coping with emergencies. Based on the NRC staff's review and documentation of requisite PSAR minimum information and evaluation provided above, the NRC staff determined that the PSAR is acceptable because it contains the necessary minimum RG 1.242 PSAR preliminary ingestion response planning design information and addresses the minimum PSAR information guidance of DANU-ISG-2022-01 and is therefore acceptable. Thus, the NRC staff concludes that the above LMGS PSAR preliminary ingestion response planning PSAR information meets the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii).

The final LMGS emergency plan ingestion response planning design details necessary demonstrate compliance with the applicable ingestion response planning criteria in 10 CFR 50.160(b)(4), would be provided by the applicant through the submittal of the final LMGS emergency plan and FSAR as part of an OL application, where the NRC staff would perform a more detailed evaluation of the LMGS facility ingestion response planning final design details under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

11.4.2 Conclusion

Based on the NRC staff's emergency preparedness review, and the determinations documented above, the NRC staff concludes that the information in LMGS PSAR section 11.4 is sufficient and, where applicable, consistent with RG 1.242 and DANU-ISG-2022-01. PSAR section 11.4, therefore, meets the regulatory requirements enumerated in 10 CFR 50.34(a)(10)(ii) and identified above in SE section 11.4.1.

11.5 Procedures

Procedures for LMGS were evaluated in section 11.1 of this SE.

11.6 Security and Fitness for Duty

The CP application for the LMGS describes the site characteristics applicable to security and provides sufficient information to demonstrate that security plans and measures will be developed in accordance with the applicable requirements of 10 CFR 73.55, "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage," and 10 CFR 100.21, "Non-seismic siting criteria." The NRC staff reviewed site-specific information in PSAR chapters 1, 2, and 11 to determine whether any characteristics of the proposed LMGS could impede the development of adequate security plans and measures. This information forms part of the basis for the NRC staff's evaluation under section 11.6 of this SE.

Site Information

PSAR chapter 1 states that the proposed LMGS site will be located on land owned by TDCC in Calhoun County, Texas (TX), approximately eight miles (12.9 km) north-northwest of Seadrift, TX. The proposed LMGS facility will be constructed adjacent to Dow SDO. PSAR figure 1.1-3 shows the site layout and PSAR figure 1.1-4 shows the plant layout.

The LMGS site consists of a nuclear island (NI) and conventional island (CI). The NI includes SSCs associated with the reactor, steam generator, fuel handling, spent fuel storage, and radwaste treatment. The CI includes SSCs associated with power production, steam export, site utilities, and power export.

The PSAR provides site information on geography and demography in section 2.1; nearby industrial, transportation, and military facilities in section 2.2; meteorology in section 2.3, hydrology in section 2.4, and geology of the site and the surrounding area in section 2.5. Information in PSAR sections 2.1 through 2.5 provides the site-specific details needed to identify potential hazards in and around the LMGS. These characteristics informed the NRC staff's review of PSAR section 11.6 by establishing whether any onsite or nearby hazards, including those associated with transportation routes, industrial or military facilities, or nearby airports, could affect the development of adequate security plans and measures in accordance with 10 CFR 100.20, "Factors to be considered when evaluating sites." PSAR section 11.6 describes site characteristics relevant to address the applicable regulatory requirements for the LMGS, in support of developing adequate security plans and measures.

PSAR figure 2.1.3-13 shows the EAB and low population zone (LPZ) and includes topographic features and transportation routes (i.e., highways, railways, and waterways) in the vicinity. No resident or transient populations exist within the EAB and LPZ. PSAR figure 2.1.3-13 also shows the EAB and LPZ overlapping segments with a calculated population of one resident. As described in PSAR section 2.1.3.1, this is an artifact of the methodology used to estimate population based on overlap between the segment and census block group.

As described in PSAR section 2.1.2 and shown in PSAR figure 2.1.2-1, the EAB and LPZ are entirely contained within the site boundary in an area controlled by the site owner. The EAB and LPZ do not contain any residences. The nearest transient facilities within 1 mi (1.6 km) of the LMGS site's 0.5 mi (0.8 km) bounding limit shown in PSAR figure 2.1.3-13 include the L&W Frazier RV Park located 0.6 mi (1.0 km) north-northeast from the bounding limit.

The provisions in 10 CFR 100.21, “Non-seismic siting criteria,” set forth the requirements regarding non-seismic siting criteria for proposed commercial power reactor sites. Specifically, 10 CFR 100.21(f) requires that site characteristics must be such that adequate security plans and measures can be developed.

The provisions in 10 CFR 73.55 require power reactor licensees and applicants to establish and maintain a physical protection program, including a security organization, which must have as its objective to provide reasonable assurance that activities involving special nuclear material are not harmful to the common defense and security and do not constitute an unreasonable risk to public health and safety.

The applicable regulatory requirements for the evaluation of the FFD and security programs are as follows:

- 10 CFR Part 26, “Fitness for Duty Programs”;
- 10 CFR 50.34(a)(1); and
- 10 CFR 100.21(f).

Acceptance criteria adequate to meet the above requirements include those set forth in:

- Regulatory Guide (RG) 4.7, Revision 3, “General Site Suitability Criteria for Nuclear Power Stations” (ML12188A053); and
- NUREG-0800 Chapter 13, section 13.6.3, Revision 2, “Physical Security – Early Site Permit and Reactor Siting Criteria” (ML14140A210). (While NUREG-0800 does not directly apply to non-LWR technologies, the characteristics of the proposed facility do not limit the NRC staff’s ability to rely on the applicable portions of this standard).

11.6.1 Technical Evaluation

11.6.1.1 Security

In conducting the technical evaluation of the information contained in PSAR chapter 11, section 11.6, the NRC staff also reviewed the pertinent information and figures contained in PSAR sections 2.1 through 2.5:

- Section 2.1, “Geography and Demography”
- Section 2.2, “Nearby Industrial, Transportation, and Military Facilities”
- Section 2.3, “Meteorology”
- Section 2.4, “Hydrology”
- Section 2.5, “Geology”

The NRC staff’s review focused on: (1) whether the information in the application meets the requirements stated in 10 CFR 50.34(a)(1)(i) to demonstrate that the site is such that security plans and measures can be developed; (2) whether the applicant has considered the applicable physical protection requirements stated in 10 CFR 73.55 in the selection of the site and its proposed layout; and (3) whether the information in the application related to the site characteristics and potential hazards provides sufficient technical basis to demonstrate that the site characteristics and potential hazards do not present impediments to preclude the development of adequate security plans and measures consistent with 10 CFR 100.21(f). These characteristics were analyzed to ensure the guidance was met in accordance with NUREG–0800, “SRP for the Review of Safety Analysis Reports for Nuclear Power Plants: Light Water Reactor (LWR) Edition.”

The NRC staff reviewed PSAR section 11.6.1, which establishes the licensing basis for demonstrating that security measures and plans can be developed. The applicant analyzed the following site characteristics: site topography and layout, transportation routes, pipelines, nearby airports and hazardous materials facilities, and other environmental features. The applicant’s evaluation of the site characteristics demonstrates that no natural or manmade site characteristics would impede the development of adequate security plans. Specifically, the flat terrain supports effective surveillance. In addition, the absence of onsite transportation routes and pipelines eliminates related security concerns, and nearby aviation activity is minimal and poses no significant hazard. Therefore, the information provided in the PSAR is sufficient for the NRC staff to conclude that the LMGS site presents no features that would hinder the establishment of an effective physical protection program. The NRC staff used review guidance in NUREG-0800, section 13.6.3, and concludes that the applicant has adequately considered site characteristics relevant to security requirements in 10 CFR Part 73. The applicant’s assessment identified no impediments to developing security plans consistent with 10 CFR 100.21(f). The NRC staff confirmed that, as no impediments were found, no mitigative actions are necessary.

PSAR section 11.6.1 states that LME, as required by 10 CFR 50.34(c) and (d), will submit the following plans as part of its application for an OL, which will meet the applicable portions of 10 CFR Part 73:

- Physical security plan
- Training and qualification plan
- Cyber security plan
- Safeguards contingency plan

The NRC staff agrees with LME that NRC regulations require these plans to be provided at the OL application stage (i.e., they are not required at the CP stage). As such, the NRC staff's evaluation of these plans would be conducted during the review of an OL application.

11.6.1.2 Fitness-for-Duty

In PSAR section 11.6.2, LME provides a description of the FFD program that would be implemented at the LMGS construction site, as required by 10 CFR 26.401(b). The NRC staff's technical evaluation consisted of a review of LME's FFD program description for consistency with the requirements in 10 CFR Part 26 that apply during reactor construction, and for consistency with the guidance contained in NUREG-0800, section 13.7.2.

Before the receipt of special nuclear material in the form of fuel assemblies, 10 CFR 26.3(c)(4) requires a 10 CFR Part 50 CP holder to implement an FFD program. Once construction activities begin, FFD program requirements apply to the individuals performing activities described under 10 CFR 26.4(e), (f), and (g).

The definition of "construction or construction activities" in 10 CFR 26.5 means "the tasks involved in building a nuclear power plant that are performed at the location where the nuclear power plant will be constructed and operated. These tasks include fabricating, erecting, integrating, and testing safety- and security-related SSCs, and the installation of their foundations, including the placement of concrete."

PSAR section 11.6.2 states that "[as] defined in 10 CFR 26.5, for the purposes of implementing 10 CFR 26, 'construction activities' are those site work tasks that include fabricating, erecting, integrating, and testing safety- and security-related SSCs, and the installation of their foundations, including the placement of concrete."

During reactor construction, any individual performing an activity described in 10 CFR 26.4(f) (i.e., constructing or directing the construction of safety- or security-related SSCs) must be subject to an FFD program that meets the requirements in 10 CFR Part 26, Subpart K, "FFD Programs for Construction." The applicant may instead choose to subject these individuals to a full FFD program (i.e., a program that meets the requirements in Subparts A-H, N, and O of 10 CFR Part 26).

PSAR section 11.6.2 states that the LMGS construction FFD program is compliant with 10 CFR Part 26 and consistent with NEI 06-06, "Fitness for Duty Program Guidance for New

Nuclear Power Plant Construction Sites," Revision 6 (ML13093A340), as endorsed, with an exception as outlined in RG 5.84, "Fitness-for-Duty Programs at New Reactor Construction Sites," Revision 0 (ML15083A412). The LMGS construction FFD program would be implemented prior to commencing construction of safety- and security-related SSCs. RG 5.84 endorsed the methodology discussed in NEI 06-06 as acceptable for implementing an FFD program at a new reactor construction site that meets the requirements of Subpart K of 10 CFR Part 26 for the worker population identified in 10 CFR 26.4(f). The NRC staff therefore determined that the FFD program description contained in PSAR section 11.6.2 defines adequate FFD program measures for the worker population identified in 10 CFR 26.4(f).

LME has requested an exemption to permit technical and vendor experts to perform any activities covered under 10 CFR 26.4(f), for a limited period of time (i.e., 30 days or less in a 60-day period), and instead of being subject to an FFD program under 10 CFR Part 26, to be treated as visitors under the control and direct observation of a trained escort who is subject to the FFD requirements of 10 CFR Part 26 (Subparts A-H, N, and O). The NRC staff's evaluation of this exemption request is contained in chapter 15 of this SE.

During reactor construction, individuals performing any applicable activity specified in 10 CFR 26.4(e) (e.g., serving as security personnel required by the NRC until receipt of special nuclear material in the form of fuel assemblies; performing quality assurance, quality control, or quality verification activities related to safety- or security-related construction activities; supervising or managing the construction of safety- or security-related SSCs) and 10 CFR 26.4(g) (e.g., FFD program personnel involved in the day-to-day operations of the FFD program, such as persons who make determinations of fitness, who notify individuals to appear for testing, or who collect specimens for drug and alcohol testing) must be subject to an FFD program that meets all the requirements in Subparts A-H, N, and O of 10 CFR Part 26. The NRC staff determined the construction FFD program for the LMGS meets the requirements of 10 CFR Part 26 based on the following:

- PSAR section 11.6.2 states that "Security personnel are covered by the construction FFD program (10 CFR 26, Subparts A-H, N, and O)." This PSAR statement is more inclusive than the duties described in 10 CFR 26.4(e)(1) which requires an individual to be subject to an FFD program if he or she "[s]erves as security personnel required by the NRC, until the licensees or other entities receive special nuclear material in the form of fuel assembles…." As a result, the NRC staff determined that the FFD program description in PSAR section 11.6.2 defines adequate FFD program measures for the worker population identified in 10 CFR 26.4(e)(1) because the PSAR is more inclusive than the duties required in the regulation.

- PSAR section 11.6.2 states that "Quality assurance, control, and verification personnel are subject to a construction FFD program (10 CFR 26, Subparts A-H, N, and O)." This PSAR statement is more inclusive than the duties described in 10 CFR 26.4(e)(2) which requires an individual to be subject to an FFD program if he or she "[p]erforms quality assurance, quality control, or quality verification activities related to safety- or security-related construction activities." As a result, the NRC staff determined that the FFD program description in PSAR section 11.6.2 defines adequate FFD program measures for the worker population identified in 10 CFR 26.4(e)(2) because the PSAR is more inclusive than the duties required in the regulation.

- PSAR section 11.6.2 states that "Construction management and oversight are covered by the construction FFD program 10 CFR 26, Subparts A-H, N, and O." This PSAR statement is more inclusive than the duties described in 10 CFR 26.4(e)(5) which requires an individual to be subject to an FFD program if he or she "[s]upervises or manages the construction of safety- or security-related SSCs." As a result, the NRC staff determined that the FFD program description in PSAR section 11.6.2 defines adequate FFD program measures for the worker population identified in 10 CFR 26.4(e)(5) because the PSAR is more inclusive than the duties required in the regulation.

- PSAR section 11.6.2 states that "Access authorization program oversight personnel, and second-level and higher supervisors and managers, are subject to a construction FFD program (10 CFR 26, Subparts A-H, N, and O)." This PSAR statement adequately aligns with 10 CFR 26.4(e)(6) which requires an individual to be subject to an FFD program if he or she "[d]irects, as defined in § 26.5, or implements the access authorization program…" As a result, the NRC staff determined that the FFD program description in PSAR section 11.6.2 defines adequate FFD program measures for the worker population identified in 10 CFR 26.4(e)(6) because the PSAR satisfies the duties required in the regulation.

- PSAR section 11.6.2 states that "FFD program personnel are covered by the construction FFD program (10 CFR 26, Subparts A-H, N and O)." This PSAR statement is more inclusive than the duties described in 10 CFR 26.4(g) which requires "[a]ll FFD program personnel who are involved in the day-to-day operations of the program, as defined by the procedures of the licensees or other entities…." to be subject to an FFD program. As a result, the NRC staff determined that the FFD program description in PSAR section 11.6.2 defines adequate FFD program measures for the worker population identified in 10 CFR 26.4(g) because the PSAR is more inclusive than the duties required in the regulation.

11.6.1.3 Access Authorization

PSAR section 11.6 provides a description of the security measures, including personnel access authorization measures, that would be implemented prior to commencing construction of safety-related and security-related SSCs. The technical evaluation for this area consisted of the NRC staff's review of the applicant's measures for consistency with guidance in NUREG-0800, section 13.6.4. Specifically, NUREG-0800, section 13.6.4 states that an access authorization program describes the requirements for deterring unauthorized activity during construction of a new reactor. Applicable program measures (as reflected in table 1 of NUREG-0800, section 13.6.4) include the verification of identity using official photo identification, conducting demographic data checks to ensure that individuals are not suspected terrorists, and providing a copy of the construction area work policy (including a description of prohibited activities, behaviors, and items) to construction personnel.

PSAR section 11.6 states that construction phase security measures, including access authorization measures, would be consistent with the guidance contained in NEI 09-01, "Security Measures During New Reactor Construction," Revision 0 (not publicly available). As discussed in SRM-SECY-10-0137, "Staff Requirements – SECY-10-0137 – Proposed Rule:

Requirements for Access Authorization and Physical Protection During Nuclear Power Plant Construction" (ML110890821), the NRC supports implementation of the personnel access authorization measures defined in NEI 09-01. The NRC staff have determined that these measures are consistent with those described in NUREG-0800, section 13.6.4 because LME has provided an access authorization program that sufficiently describes the requirements for deterring unauthorized activity during construction, and therefore the NRC staff concludes that PSAR section 11.6.2 describes adequate access authorization measures.

11.6.2 Conclusion

The NRC staff concludes that the applicant provided sufficient technical basis to demonstrate that the site characteristics and potential hazards do not present impediments that would preclude the development of adequate security plans and measures. The NRC staff also concludes that the LMGS site is such that adequate security plans and measures can be developed in accordance with the requirements in 10 CFR 50.34(a)(1)(i) and 10 CFR 100.21(f).

11.7 Technical Specifications

Part IV of the LMGS CP application provides an identification and justification for the selection of variables, conditions, or other items which are determined as the result of the preliminary safety analysis and evaluation to be the probable subjects of technical specifications (TSs) for the facility, with special attention given to those items which may significantly influence the final design.

LME provides operating modes in table IV-1 of the CP application. These are unit operating modes that correspond to an SSCs configuration, average primary pressure, and fuel pebble status in the reactor vessel. They include the following:

- Mode 1: Power Operation
- Mode 2: Startup
- Mode 3: Shutdown
- Mode 4: Depressurized
- Mode 5: Defueled

Preliminary safety limits are defined in table IV-2 of the CP application and include specified acceptable radionuclide release design limits (SARRDLs), reactor pressure vessel (RPV) pressure, and RPV temperature. LME states that limiting safety system settings, analytical limits, and setpoints to ensure safety limits are not exceeded will be provided at the OL application stage. Table IV-3 of the CP application contains a list of systems and variables that are the preliminary subjects for limiting conditions of operation (LCOs). LME used the four criteria provided in DANU-ISG-2022-08 to determine whether an SSCs will be the subject of an LCO. LME plans to use the format in NUREG-1431, Volume 1, Revision 5 for LCOs provided at the OL application stage.

Part IV of the CP application also provides a description of the preliminary proposed design features to be included in the TSs and a preliminary description of the areas to be addressed with administrative controls at the OL application stage.

The applicable regulatory requirements for the evaluation of the TSs are as follows:

- 10 CFR 50.34(a)(5).

The applicable guidance for the evaluation of the TSs are as follows:

- DANU-ISG-2022-08, "Risk-Informed Technical Specifications" (ML23277A146); and
- NUREG-1431, Volume 1, Revision 5, "Standard Technical Specifications" (ML21259A155)

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11.7.1 Technical Evaluation

The regulations in 10 CFR 50.34(a)(5) require a CP applicant to identify and justify the selection of those variables, conditions, or other items which are determined as the result of preliminary safety analysis and evaluation to be probable subjects of TSs for the facility, with special attention given to those items which may significantly influence the final design. Acceptance criterion 1 from DANU-ISG-2022-08 specifies that the reviewer should be able to reach a safety finding and address the topic(s) in the NRC's SE report if the application includes the information required by 10 CFR 50.34(a)(5).

Part IV of the CP application states that the format and content of the LMGS TSs, which would be submitted with a LMGS OL application, would:

1. address the requirements in 10 CFR 50.36 in accordance with the 10 CFR 50.34(b)(6) requirement that TSs be included in an OL application; and
2. be consistent with guidance provided in NUREG-1431, "Standard Technical Specifications," Revision 5, Volume 1.

Part IV, section 1.1 of the CP application describes the five operational modes for the LMGS TSs, noted above, which are defined in terms of combinations of control rod and shutdown rod location, primary loop pressure, and fuel loading. These operational modes are acceptable for the CP because they are consistent with the operational modes used in the preliminary safety analysis, documented in PSAR table 3.11-1. See section 3.11 of this SE for additional discussion on operational modes.

CP application table IV-2 indicates that safety limits (SLs) will be placed on certain process variables, specifically, SARRDLs, RPV pressure, and RPV temperature. While LME has not identified peak fuel temperature as its own SL, the SLs associated with not exceeding the SARRDLs and the RPV temperature are inherently tied to fuel temperatures and will likely require limits on parameters such as peak helium temperature during normal operation to ensure these SLs are not exceeded during any applicable LBE transients.

CP application table IV-3 provides a preliminary list of systems that will be subject to LCOs, along with variables and conditions that the applicant expects to be subject to TSs, based on its preliminary safety analysis described in the PSAR. CP application table IV-3 also summarizes LME's basis for each listed TS variable or condition. The systems included in the proposed preliminary LCOs and the corresponding SE section where they are evaluated are as follows:

- Reactor protection system (6.4.13)
- Nuclear instrumentation system (6.4.14)
- Investment protection system (7.3.11)
- Steam generator system (6.4.12)
- Reactivity control and shutdown system (6.4.10)
- Reactor pressure vessel (6.4.6)
- Defuel chute pressure vessel (6.4.8)
- Primary pressure relief system (6.4.9)
- Reactor building (6.4.1)
- Helium circulator system (7.3.5)
- Fuel handling system (7.3.6)

Part IV, section 2.5 of the CP application provides a list of probable design features, which, if altered or modified, could have a significant effect on safety. These probable design features include the RPV, graphite core assembly, core metallic structures, defuel chute, fuel and functional containment, and the spent fuel storage system. Part IV, section 2.6 of the CP application lists probable administrative controls relating to organization and management, procedures, and reporting necessary to operate the plant in a safe manner. These probable administrative controls include the operations organization structure (including roles and responsibilities, facility staff qualifications, as well as requirements to establish, implement and maintain procedures, programs, and reports).

The NRC staff evaluated the sufficiency of the preliminary information on LME's TSs, as described in CP application Part IV, against the requirements of 10 CFR 50.34(a)(5). In its evaluation, the NRC staff also considered the PSAR information and discussions of the probable subjects of TSs in other PSAR chapters. The NRC staff finds that LME's probable subjects of TSs are consistent with important parameters determined as a result of the preliminary safety analyses in the PSAR, based on the system evaluations listed above and with special attention given to items (including SLs and LCOs) that may significantly influence the final design, and that the probable subjects of TSs are supported by appropriate bases. Therefore, based on the information in CP application Part IV and other PSAR chapters, the NRC staff finds that LME's identification and justification of the preliminary subjects of TSs meets the requirements of 10 CFR 50.34(a)(5). The NRC staff will perform a detailed evaluation of the complete and finalized TSs for LMGS, including SLs, LCOs, design features, administrative controls, and operational modes during its review of any OL application.

11.7.2 Conclusion

Based on the evaluations described above, the NRC staff concludes that the information in Part IV of the CP application is sufficient, is consistent with the applicable guidance, and meets the regulatory requirements in 10 CFR 50.34(a)(5).

12 INITIAL TEST PROGRAM

Preliminary Safety Analysis Report (PSAR) chapter 12 describes the initial test program (ITP) that is performed during the initial startup of the Long Mott Generating Station (LMGS).

The applicable regulatory requirements for the evaluation of the ITP are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a)(7); and
- Criterion XI, "Test Control," of Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

The U.S. Nuclear Regulatory Commission (NRC) staff has issued the following guidance for the development and review of the post-construction inspection, testing, and analysis program (PITAP):

- Division of Advanced Reactors and Non-Power Production and Utilization Facilities (DANU) Interim Staff Guidance (ISG) DANU-ISG-2022-06, "Advanced Reactor Content of Application Project, Chapter 12 'Post-manufacturing and construction Inspection, Testing, and Analysis Program,'" (ML23277A144).

12.1 Technical Evaluation

The NRC staff reviewed the LMGS ITP against the criteria provided in DANU-ISG-2022-06. For a construction permit (CP) application, DANU-ISG-2022-06 states that the PITAP description can be limited to the preoperational phase inspection, testing, and verification that would be required by 10 CFR Part 50, Appendix B, and that the PITAP should include a description of the scope, objectives, and programmatic controls associated with the preoperational test program (prior to initial fuel loading). Accordingly, the NRC staff's review, as presented below, is limited to the: (1) scope, (2) objectives, and (3) programmatic controls for Phase 1 of the LMGS ITP (i.e., preoperational inspection, testing, and verification).

12.1.1 Scope

The NRC staff evaluated the structures, systems, and components (SSCs) included in the scope of the LMGS ITP cold commissioning and preoperational testing consistent with DANU-ISG-2022-06, section F, "Acceptance Criteria," which states that the preoperational inspection, testing, and verification program (including elements of the quality assurance program (QAP), as applicable) includes all safety-related (SR) and safety-significant SSCs that can reasonably be verified at the preoperational stage.

The NRC staff reviewed PSAR section 5.1, which states: "Safety-significant SSCs include all those SSCs classified as SR or NSRST [non-safety-related with special treatment]." PSAR section 12.1.1 describes the commissioning process and states that the scope of the ITP addresses all safety-significant SSCs.

The NRC staff determined that the description of the scope provided in the PSAR is consistent with DANU-ISG-2022-06 and, therefore is acceptable because the ITP addresses all safety-significant SSCs, which includes all SSCs classified as SR or NSRST.

12.1.2 Objectives

The NRC staff reviewed the ITP objectives consistent with DANU-ISG-2022-06, which states that the primary objective of the PITAP is to demonstrate, to the extent possible, that the safety-significant SSCs as constructed or manufactured, as the case may be, will operate in accordance with the design and as described in the safety analysis report.

DANU-ISG-2022-06 also states that additional objectives of the PITAP for preoperational testing include:

- providing reasonable assurance that the facility exhibits the performance and associated safety margins described in the design;
- satisfying any license conditions associated with the PITAP;
- obtaining as-built data to verify the analytical assumptions, limits, and models;
- familiarizing the plant's operating and technical staff with operation of the facility; and
- verifying the adequacy of the plant operating and emergency procedures.

The NRC staff did consider objectives in DANU-ISG-2022-06 that are not applicable to CP applications.

The NRC staff reviewed PSAR section 12.1.1, which states that: "[t]he objective of the commissioning strategy is to demonstrate that the plant as constructed meets the design and safety requirements specified in the Final Safety Analysis Report (FSAR), license conditions, Technical Specifications (TS), and as specified in the requirements management program through corresponding acceptance criteria." To accomplish this, PSAR section 12.1.1 also states that the LMGS ITP includes:

- Verification that safety-significant SSCs fulfill design and safety requirements through attainment of corresponding acceptance criteria.
- Measurement and collection of plant operating data for SSCs and integrated plant responses for validation of analyses and future reference as well as to validate any applicable analytical assumptions, limits, or models.
- Validation that all operating, surveillance, and emergency procedures are adequate.
- Demonstration of operating staff proficiency and, that maintenance and technical staff have the required skills and abilities to fulfill their job functions.

PSAR chapter 12 also states that the ITP provides assurance that the operating and emergency procedures have been evaluated and demonstrated, and that the plant operations personnel are knowledgeable about the plant operating procedures and are fully prepared to operate the plant in a safe manner.

In PSAR section 12.1.2, Long Mott Energy, LLC., described the scope of first-of-a-kind (FOAK) SSC testing that will be performed under the ITP. As stated in PSAR section 12.1.2, the tests

are intended to “verify that safety-significant FOAK SSCs not validated through other verification methods were constructed and will operate in accordance with their design.” The NRC staff determined that the preliminary description in PSAR section 12.1.2 meets the acceptance criteria by identifying and including FOAK SSCs in the ITP.

The NRC staff determined that the PSAR description captures those portions of the ITP objectives described in DANU-2022-06 that are relevant at the CP stage. Further, nothing about the differences between the submitted ITP description and the guidance in DANU-ISG-2022-06 necessitates a different or expanded ITP objective. This is because the ITP objectives described in the PSAR capture what is necessary to meet the applicable regulatory requirements for issuance of a CP in accordance with 10 CFR 50.34(a) and 10 CFR Part 50, Appendix B. Therefore, the NRC staff finds that the objectives of the LMGS ITP are acceptable.

12.1.3 Programmatic Controls

The NRC staff reviewed the ITP cold commissioning testing and preoperational testing programmatic controls described in the PSAR. PSAR chapter 12 states that the ITP is executed under the QA program (QAP) described in the topical report, “Long Mott Energy, LLC Quality Assurance Program Description” (ML24345A045). The NRC staff’s review of the QAP as applicable to the ITP is discussed in section 8.1 of this safety evaluation (SE).

Based on the use of an acceptable QAP, as described in section 8.1 of this SE, and because the LMGS QAP description contains adequate programmatic controls for the conduct of the ITP, the NRC staff determined that the programmatic controls for conduct of the LMGS ITP are acceptable.

12.2 Conclusion

Based on the information above, the NRC staff finds that the ITP is acceptable for addressing the test control requirement under 10 CFR 50.34(a)(7) and Criterion XI, “Test Control,” of Appendix B to 10 CFR Part 50.

13 RESEARCH AND DEVELOPMENT

Chapter 13, “Research and Development Program,” of the Long Mott Generating Station (LMGS) preliminary safety analysis report (PSAR) identifies those structures, systems, and components (SSCs) that require further research and development (R&D) to confirm their design requirements. The PSAR states that the R&D program is designed to resolve outstanding safety questions necessary to confirm the adequacy of the design of certain SSCs prior to completion of construction of LMGS.

The applicable regulatory requirements for the evaluation of R&D are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a)(8);
- 10 CFR 50.35(a)(2);
- 10 CFR 50.35(a)(3);
- 10 CFR 50.35(b); and
- 10 CFR 50.43(e).

The NRC staff notes that while 10 CFR 50.43(e) is not applicable to a construction permit (CP) application, completion of the R&D activities discussed in the PSAR may, in part, be necessary for the required performance demonstrations at the operating license (OL) stage. As such, the U.S. Nuclear Regulatory Commission (NRC) staff considered the R&D activities in contemplation of ensuring a clear path to demonstrating compliance with 10 CFR 50.43(e) at the OL phase.

Applicable guidance on the NRC staff’s review of R&D activities includes:

- Division of Advanced Reactors and Non-Power Production and Utilization Facilities (DANU) interim staff guidance (ISG) DANU-ISG-2022-01, “Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications - Roadmap,” Appendix C (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23277A139).

13.1 Technical Evaluation

13.1.1 R&D Program Definition and Scope

The NRC staff reviewed PSAR section 13.2, which states that the scope of the LMGS R&D program is determined based on an SSC demonstrating consistency with the following criteria:

- the subject activity must meet the 10 CFR 50.2, “Definitions, definition of R&D, and
- the subject SSC must fall within the established 10 CFR 50.34(a)(8) scope of an R&D program by having an outstanding safety question that must be resolved to confirm the adequacy of the design by the completion of construction.

The PSAR further cites the definition of R&D from 10 CFR 50.2 and states that LME has not identified any SSCs or activities that would be appropriately characterized as needing "research" to resolve outstanding safety questions tied to the construction of LMGS. LME characterizes any such needs as "development" related. The PSAR defines "research" activities as "focused on experimental or theoretical work that may not have a defined use case" and "development" activities as "apply the findings of previous investigative work to specific testing needed to confirm the adequacy of the LMGS design," consistent with 10 CFR 50.2.

Further, PSAR figure 13.2-1 lists being an SSC as a key criterion for inclusion in the R&D program. As such, the PSAR narrows down the criteria for inclusion into the LMGS R&D program to: (1) the subject activity meeting the 10 CFR 50.2 definition of R&D, (2) the subject of the activity being an SSC, and (3) the subject SSC falling within the scope of an R&D program, as described in 10 CFR 50.34(a)(8).

While 10 CFR 50.34(a)(8) focuses the scope of R&D on SSCs, the NRC staff's view is that the intent of 10 CFR 50.34(a)(8), in consideration of the definition of R&D in 10 CFR 50.2, is not to limit R&D to only physical aspects of the SSC, but rather what is needed to understand the SSCs and their functions to the level that the safety analysis can be confirmed to be valid, which is the basis for the SSC classification, capability targets, and special treatment needed to set design requirements impacting the physical SSCs. For example, the transient and safety analysis methodology is a model used to represent the expected performance of the physical SSCs within the facility during a transient event. If additional research is needed to provide information to accurately model the safety-significant SSCs (e.g., reactor cavity cooling system testing) this would be within the definition of 10 CFR 50.2 and the scope of 10 CFR 50.34(a)(8), and LME's process for reviewing and updating the R&D program would result in these SSCs being scoped into the R&D program.

The scope of the LMGS R&D program includes safety-related (SR) SSCs and non-safety-related with special treatment (NSRST) SSCs that perform risk-significant functions but excludes NSRST SSCs that are only required for defense-in-depth (DID) adequacy. The PSAR states that this is appropriate because the NSRST SSCs relied upon for DID adequacy are not required to prevent or mitigate any licensing basis event (LBE) from exceeding the frequency-consequence (F-C) curve and do not provide a significant contribution to the cumulative risk metrics selected for the total risk from all analyzed LBEs. While these statements accurately align with the LMP SSC classification definitions in Nuclear Energy Institute (NEI) 18-04, "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development," LMP also emphasizes the importance of maintaining adequate DID within the facility to ensure safety. The NRC staff does not see any language in 10 CFR 50.2 or 10 CFR 50.34(a)(8) that would indicate an intent to exclude these SSCs from the scope of what is considered for R&D.

While the methodology used to scope activities and SSCs into LME's R&D program appears restrictive, the NRC staff confirmed in regulatory audit (ML26293A340) that testing or further development needed to resolve safety aspects for activities and SSCs that are outside the scope of the R&D program will still be performed as described in the PSAR and as needed to support the OL application. LME indicated that the R&D program is a licensing program used to track the activities and SSCs that LME has identified as falling into 10 CFR 50.34(a)(8). For example, the full scope of the materials qualification described in PSAR section 3.8 will be performed under a program designed to ensure material aspects that could impact

safety-significant SSCs are confirmed prior to the OL application. However, only those items related to SR and risk-significant NSRST SSCs will be designated in the R&D program.

The PSAR further states that the R&D program will be reviewed and updated on an ongoing basis and that SSCs may be added or removed from the LMGS R&D program: a) if the SSC classifications are changed based on the PRA, and/or b) if the safety questions associated with an R&D item are either identified or eliminated as part of the ongoing updates. The PSAR states that for items in the R&D program where safety questions are resolved, the activities will be marked complete in the R&D program.

As captured in RCI-1 (ML26239A157) and confirmed by LME, additional risk-significance evaluations performed indicate that multiple systems currently classified as NSRST for DID may be re-classified as risk-significant NSRST. Based on the methodology defined in PSAR chapter 13, these SSCs will be re-evaluated by LME for inclusion in the R&D program when changes to the safety classification or risk-significance occur.

13.1.2 SSCs in the LMGS R&D Program

This section presents the NRC staff's evaluation of the items identified by LME as within the R&D program.

Graphite Core Assembly

The NRC staff reviewed PSAR section 13.3.1, which indicates that uncertainties exist around the performance of Graphite material in the Xe-100 operating environment. The Xe-100 operating environment is expected to be a high fluence, high temperature, and oxidizing environment. The uncertainties are primarily due to limitations of existing codes and standards. Therefore, LME included the graphite core assembly (GCA) in its R&D program to resolve outstanding questions regarding the GCA's performance in a high fluence, high temperature, and oxidizing environment which can potentially impact its ability to perform its required safety functions (RSFs).

The PSAR states that R&D testing is described in detail in "Xe-100 Licensing Topical Report Graphite Core Assembly Material Qualification and Design Methodologies, Revision 2" (ML26114A162, incorporated by reference) and that any outstanding safety questions will be addressed prior to completion of construction of LMGS. PSAR table 1.4-14 lists the limitations and conditions (L&Cs) associated with use of the Licensing Topical Report (LTR) along with LME's proposed plan to meet each condition as part of the R&D program. The acceptability of LME's proposed plan to address L&Cs associated with the use of GCA LTR as part of R&D program is discussed in section 1.4.1 of this SE.

Because further testing performed for the GCA in a prototypic Xe-100 operating environment would reasonably resolve the safety questions related to its performance in a high fluence, high temperature, and oxidizing environment, and because the work will be completed prior to the completion of construction, the NRC staff determined that the R&D activities planned for GCA are consistent with the requirements of 10 CFR 50.34(a)(8) and 10 CFR 50.35(a)(3).

Safety-Related Metallic SSCs

The NRC staff reviewed PSAR table 13.3-1, which lists the metallic assessments, testing, and analysis and the affected SR metallic SSCs that are included in the R&D program. PSAR section 13.3.2 indicates that uncertainties exist around the performance of metallic materials of these SSCs in the Xe-100 operating environment primarily due to limitations of existing codes and standards. The PSAR states that no additional code and environmental assessments tied to SR SSCs, beyond those listed in PSAR table 13.3-1, have been identified as necessary for the preliminary design.

The PSAR states that R&D testing is described in detail in the LMGS material qualification plan and any outstanding safety questions identified will be addressed prior to completion of the construction of LMGS. The acceptability of the LMGS material qualification plan is discussed in section 3.8 of this SE. As confirmed by the NRC staff in RCI-1 (ML26239A157), LME is in the process of incorporating the results of updated risk-significance evaluations following the NEI 18-04 methodology. As stated in PSAR section 13.3.2, SR and risk significant NSRST SSCs may be added to the R&D program based on changes in classification. For the SSCs that elevate to risk-significant based on the evaluation described in RCI-1, and have outstanding safety questions to be addressed, the SSCs would be added to the R&D program based on the LME methodology.

Because LME's planned testing for the SR metallic SSCs is in a prototypic Xe-100 operating environment (i.e., high fluence, high temperature, and oxidizing environment), the NRC staff determined that the planned testing will reasonably resolve the associated safety questions. Additionally, the work is expected to be completed prior to completion of the construction of LMGS, in alignment with 10 CFR 50.34(a)(8) requirements. Therefore, the NRC staff determined that the R&D activities planned for SR metallic SSCs are consistent with the requirements of 10 CFR 50.34(a)(8) and 10 CFR 50.35(a)(3).

Fuel

The NRC staff reviewed PSAR section 13.3.3, which includes Xe-100 fuel as part of the LMGS R&D program to confirm acceptable burnup, fast fluence, and temperature limits planned for the operation of Xe-100. The planned testing is to confirm the gaps in safety testing to ensure that the burnup, fast fluence, and temperature limits outside the validated range for these parameters will not challenge the Tri-Structural Isotropic (TRISO) fuel particles and pebbles to perform its RSF. The PSAR states X-energy Pebble Reactor Test (XPeRT) irradiation testing is planned, as described in PSAR section 6.4.2.3, to demonstrate performance of the Xe-100 fuel.

The NRC staff reviewed the fuel qualification methodology elements discussed in PSAR section 6.4.2.3 and finds that the elements include confirmation of gaps in the safety testing to validate burnup, fast fluence, and temperature ranges for the Xe-100 fuel. Specifically, the testing program supports the evaluation models to conservatively predict fission product release from fuel pebbles under operating and licensing basis event conditions.

The NRC staff notes that preliminary peak fuel temperatures for the limiting DBA analyzed in PSAR section 3.6 are beyond the limits typically associated with TRISO fuel particles, including the results from the Department of Energy's (DOE) Advanced Gas Reactor (AGR) program. A survey of literature and testing data available for TRISO fuel ("TRISO Fuel: Properties and Failure Modes," ML21175A1522) generally indicates that modern TRISO fuel demonstrates

robustness through 1600°C) (2912 degrees Fahrenheit (°F)), with 1800°C (3272°F) generally being best characterized as the severe accident temperature. The NRC staff finds that planned testing to specifically qualify the fuel for temperatures expected during LBEs for Xe-100, including accounting for uncertainties, is an essential step to resolving safety questions pertaining to retention of fission products in fuel particles and pebbles, and overall release.

Because LME's planned testing addresses burnup, fast fluence, and temperature ranges for the Xe-100 fuel, the NRC staff determined that the planned testing will reasonably resolve the associated safety questions. Additionally, the work is expected to be completed prior to completion of the construction of LMGS, in alignment with 10 CFR 50.34(a)(8) requirements. Therefore, the NRC staff determined that the R&D activities planned for the fuel qualification methodology are consistent with the requirements of 10 CFR 50.34(a)(8) and 10 CFR 50.35(a)(3).

13.2 Conclusion

The NRC staff finds that LME meets the requirements of 10 CFR 50.34(a)(8) because LME has: (1) identified the SSCs that require research and development to confirm the adequacy of the design, (2) identified and described the R&D program which will be conducted to resolve any safety questions associated with such SSCs, and (3) has provided a commitment that these safety questions will be resolved prior to completion of construction of the LMGS.

Based on these considerations, the NRC staff determined that the discussion presented in the R&D program is sufficient to support issuance of a CP pursuant to the requirements of 10 CFR 50.35(a)(3).

14 FINANCIAL QUALIFICATIONS

This chapter of the safety evaluation (SE) describes the U.S. Nuclear Regulatory Commission (NRC or Commission) staff's review and evaluation of Long Mott Energy, LLC's (LME) financial qualifications as presented in Part I, "General and Financial Information," of revision 1 of the Long Mott Generating Station (LMGS) construction permit (CP) application (Agencywide Documents and Access Management System (ADAMS) Accession No. ML26261A423), as well as LME's requested exemption from the financial qualification requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.33(f) and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix C, "A Guide for the Financial Data and Related Information Required to Establish Financial Qualifications for Construction Permits and Combined Licenses" in Part V, "Non-Applicabilities and Exemptions," of the CP application (ML26261A427). In a letter dated March 3, 2026, LME submitted a proprietary supplement to Part I of the CP application (ML26062A842), which was included in revision 1 of the CP application (ML26261A419). With certain exceptions, a Part 50 application is required to provide sufficient information to demonstrate to the Commission the financial qualification of the applicant to carry out, in accordance with regulations in 10 CFR Part 50, the activities for which the permit or license is sought. The applicable regulatory requirements for the evaluation of the LME's financial qualifications are as follows:

- 10 CFR 50.33, "Contents of applications; general information,"
- 10 CFR Part 50, Appendix C,
- 10 CFR 50.38, "Ineligibility of certain applicants,"
- 10 CFR 50.54, "Conditions of licenses,"
- 10 CFR 70.23, "Requirements for approval of applications," and
- 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements."
- The applicable guidance for the evaluation of LME's financial qualifications is: DANU-ISG-2022-01, section 16, "Financial Qualification and Insurance and Liability."

In addition, the NRC staff considered the Financial Qualifications for Reactor Licensing Rulemaking dated November 16, 2016 (ML15322A185). The NRC staff evaluated the financial qualifications in section 14.1 and the exemption request in section 14.2 of this SE chapter[1].

[1] By memorandum dated June 25, 2026, the NRC's Atomic and Safety Licensing Board denied San Antonio Bay Estuarine Waterkeeper's (intervenor) contentions associated with this review area (ML26176A482).

14.1 Financial Ability to Construct LMGS

14.1.1 Construction Costs

Technical Evaluation

The NRC staff reviewed the projected costs for the construction of the proposed LMGS facility provided in CP application part I, table I-1, "Estimates of Long Mott Generating Station Construction Costs (2024 dollars)." The total estimated projected costs include estimates for total nuclear production plant costs, transmission, distribution, general plant costs, and nuclear fuel inventory cost for the first core for each of the four reactor modules comprising the LMGS design.

The NRC staff evaluated the sufficiency of the information provided for the LMGS construction and related fuel cycle costs. The NRC staff evaluated the sufficiency of this information against the financial qualification standard under 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," based on the exemption request that is evaluated in SE section 14.2.

The NRC staff's review of the application identified that LME's construction cost estimate included a detailed breakdown of the cost elements including the factors related to facility preparation and development, equipment and supplies, and labor (corporate and construction) factors. Additionally, the cost estimate was developed with support from the Dow Chemical Company (TDCC), which is LME's parent company. Based on the NRC staff's review of LME's construction cost estimate in section 2.1 of Part I of the CP application, the NRC staff determined that LME used an appropriate approach to assess the size and scope of construction. Additionally, the NRC staff reviewed LME's relationship with TDCC. Further, based on the information provided in Part I of the CP application, which includes examples of active and completed "large-build megaprojects" managed by TDCC, the NRC staff determined that TDCC has significant experience in capital sourcing, designing, procuring, and implementing "large-build megaprojects" globally. This gives the NRC staff additional confidence that the projected construction cost values are reasonable.

Based on the NRC staff's evaluation of the cost estimates for LMGS and its relationship with TDCC, the NRC staff determined that LME's cost estimate is reasonable.

14.1.2 Financial Qualifications

Consistent with the NRC staff's determination on the requested exemption as discussed in SE section 14.2, the NRC staff evaluated LME's financial qualifications using the standard described in 10 CFR 70.23(a)(5). Under this requirement, the NRC staff must determine whether the applicant "appears to be" financially qualified to engage in the proposed activities.

Technical Evaluation

The NRC staff reviewed part I of the LME CP application and evaluated whether LME meets the financial qualification requirement in 10 CFR 70.23(a)(5). As discussed in section 14.2 of this SE, the NRC staff determined that an exemption from the CP financial qualification requirements in 10 CFR 50.33(f) and 10 CFR Part 50, Appendix C, is warranted, as requested in Part V – Non-Applicabilities and Exemptions of the CP application.

Appendix A to part I of the CP application contains a financial capacity plan (FCP) which includes a construction cost estimate. The FCP demonstrates LME's understanding of the size and scope of construction of LMGS, including the level of capital necessary to undertake the project. The FCP also discusses the organizational and human resources, experience, skills, and expertise required to obtain proper financing and ultimately finance the project.

The NRC staff evaluated LME's demonstration of reasonable assurance that it will obtain funding for construction that is equal to or more than 50 percent of the LMGS construction cost estimate. To determine if the applicant "appears to be" financially qualified, the NRC staff considered section 7.2.1.2 of the regulatory basis for the Financial Qualifications for Reactor Licensing Rulemaking. In reviewing an applicant's financial qualifications, the NRC staff recognizes the committed sources of funding described in an application as contributing to financial qualification requirements and considers applicants that have greater than 50 percent committed funding sources at the time of the application to appear to be financially qualified.

The NRC staff evaluated the "Financial Support Agreement Between Dow Chemical Company and Long Mott Energy, LLC" (FSA) dated February 23, 2026, that established that LME's parent company, TDCC, will provide financial support to LME for construction of the LMGS (ML26062A842). The financial support agreement was incorporated in the revision 1 of the CP application and includes the parent company's agreement to provide funding for construction costs up to a specified dollar amount unless the parent company procures a substitute loan facility and/or a letter of credit that meets the NRC's financial assurance requirements and obtains the NRC's written consent for such substitution. The amount of committed funding is more than 50 percent of the projected construction cost estimate for the LMGS. This financial support agreement provides a mechanism for LME to incur project costs prior to receiving federal reimbursement through the U.S. Department of Energy's (DOE) Advanced Reactor Demonstration Program (ARDP). Based on the committed funding provided through the FSA between TDCC and LME, the NRC staff has determined that LME demonstrates that it appears to be financially qualified by establishing that it has secured greater than 50 percent of the required construction funding for the LMGS.

Accordingly, based on its review of the FCP and the FSA, the NRC staff determined that the level of detail provided by LME related to its understanding of the size and scope of constructing the LMGS, its parent company's experience in other large-scale projects such as the examples provided in part I of the CP application, and the company's agreement to provide 50 percent or more of the estimated cost of construction is sufficient to meet 10 CFR 70.23(a)(5)[2]. Therefore, based on these considerations including the current level of committed funds, the NRC staff finds that LME "appears to be financially qualified" to construct the LMGS.

14.1.3 Conclusion

Based on the NRC staff's conclusions above, the NRC staff finds that the information provided by LME regarding its financial ability to construct the LMGS is sufficient and that LME meets the 10 CFR Part 70 standard of "appears to be financially qualified" for the proposed activities.

[2] The NRC's Atomic Safety and Licensing Board reached a similar conclusion in *Long Mott Energy, LLC* (Long Mott Generating Station), LBP-26-4 (June 25, 2026), *appeal pending,* (dismissing Contention 3B and finding that LME had demonstrated at least 50 percent of the funding required to construct LMGS).

14.2 Request for Exemption Regarding Financial Qualifications

In part V of the CP application (ML26261A427), LME requested an exemption under 10 CFR 50.12(a)(2)(ii) and 10 CFR 50.12(a)(2)(vi) from the financial qualification requirements in 10 CFR 50.33(f), and 10 CFR Part 50, Appendix C. Per the applicant's submittal, under this exemption request the NRC would apply the financial qualification standard required for 10 CFR Part 70 applicants (for licenses to receive title to, own, acquire, deliver, receive, possess, use and transfer special nuclear material) as described in 10 CFR 70.23(a)(5), in lieu of the financial qualification standards in 10 CFR 50.33(f), and 10 CFR Part 50, Appendix C. Specifically, LME requests that it be permitted to demonstrate that it appears to be financially qualified for the issuance of a CP rather than being required to meet the requirements of 10 CFR 50.33(f) and 10 CFR Part 50, Appendix C.

Issuance of this exemption would, in effect, allow the use of the 10 CFR Part 70 financial qualification standard, normally used for licensing the possession and use of special nuclear material, for the construction of the LMGS.

14.2.1 Technical Evaluation

The requested exemption would exempt LME from the financial qualification requirements in 10 CFR 50.33(f) and 10 CFR Part 50, Appendix C. As stated above, 10 CFR 50.12(a) allows the NRC to grant exemptions from regulations should certain criteria be met. This exemption evaluation focuses on LME's proposal to use the 10 CFR Part 70 standard of "appears to be financially qualified." This section of the SE describes the NRC staff's evaluation of the exemption request.

As further discussed below, the NRC staff concludes that LME's exemption request satisfies the following provisions in 10 CFR 50.12(a):

<u>The Exemption is Authorized by Law</u>

The Commission has the authority to issue LME's requested exemption. The exemption would not conflict with any provision of the Atomic Energy Act (AEA) of 1954, as amended, or any other law. In particular, the NRC has broad discretion to prescribe requirements for financial qualifications, as provided in section 182(a) of the AEA. As such, the requested exemption is authorized by law.

<u>The Exemption Presents no Undue Risk to Public Health and Safety</u>

The NRC staff evaluated LME's request for an exemption from the requirements of 10 CFR 50.33(f) and 10 CFR Part 50, Appendix C, to determine if it presents an undue risk to public health and safety. To support the NRC staff's evaluation, the NRC staff reviewed information provided by the applicant in part V of the CPA, which states that the exemption does not pertain to any NRC safety requirements that directly govern design or construction activities for the LMGS and the use of the 10 CFR 70 financial qualification standards for LMGS construction would not compromise public health and safety. Additionally, the applicant referenced the NRC's regulatory basis document for the proposed amendment to the financial requirement in 10 CFR Part 50 (ML15322A185), which explains how the agency's oversight programs ensure public health and safety and states that an applicant demonstrating

commitments for at least 50 percent of construction funding provides a sufficient demonstration of financial capacity.

In an NRC staff requirements memorandum (SRM) titled "Policy Options for Merchant (Non-Electric Utility) Plant Financial Qualifications" (SRM-SECY-13-0124), the Commission approved the NRC staff's recommendation to conduct a rulemaking and directed the NRC staff to amend the financial qualification requirements in 10 CFR Part 50 to conform to the financial qualification review standard in 10 CFR Part 70 (ML14114A358). In response to the SRM, the NRC staff developed a proposed rule; in the regulatory basis for the proposed Financial Qualifications for Reactor Licensing Rulemaking (ML15322A185), the NRC staff stated that:

> The NRC staff believes that an applicant with commitments for at least 50 percent of its construction funding has made a sufficient demonstration of financial capacity. … Accordingly, where the applicant has identified such commitments, a license condition requiring documentation for the remaining portion of the construction funding is not necessary. This is because the purpose of the NRC staff's review is not to ensure that the project is completed; rather, it is to ensure that an applicant has the financial capacity to obtain financing when the project moves forward.

In addition, SECY-13-0124, "Policy Options for Merchant (Non-Electric Utility) Plant Financial Qualifications," (ML13057A006) states the following:

> Commission history and precedent has consistently shown an ongoing concern for the potential of degraded safety in the face of degraded financial qualifications. However, this history also consistently indicates a Commission belief that any nexus between safety and the NRC's review of financial qualifications is indirect and of secondary importance to ensuring public health and safety. Furthermore, it is clear that Section 182a. of the AEA grants the Commission much flexibility in the area of financial qualifications requirements.

Additionally, as discussed in SECY-79-299, "Generic Issue of Financial Qualifications (ML12236A723), the Commission previously concluded that financial qualification reviews are only one part of the NRC's safety oversight of nuclear power reactor applications. As the Commission observed,

> although technical reviews and inspection efforts are very effective direct methods of discovering deficiencies that could affect safety, the analysis of financial qualifications is an additional method, albeit indirect, of determining an applicant's ability to satisfy safety requirements. The financial qualifications review is one element in the Commission's system of multiple and redundant safety reviews and inspections. The purpose of the financial qualifications review in this system is analogous to the overlapping protective echelons of the 'defense-in-depth' approach used in designing nuclear power plants.

These regulatory program assessments underscore the indirect effect resulting from the financial qualifications review, as compared to more direct safety evaluations, in assuring the safety of a proposed facility. Further, the NRC staff notes that the direct means to ensure safety include a detailed technical licensing review of CP applications, the construction reactor

oversight process, the vendor inspection program, and the quality assurance inspection program.

Further, various programs described in the CP application, support a determination as to whether the LMGS will be safely constructed. These programs include the LMGS quality assurance program and the reliability and integrity management program which are described in section 8.1 and section 8.2 of the Preliminary Safety Analysis Report, respectively and are evaluated in sections 8.1 and 8.2 of this SE. Additionally, the applicant used the licensing modernization project (LMP) methodology, which provided a structured approach to development of the licensing basis that enables a comprehensive understanding of the risk and safety-significance of the facility's structures, systems, and components (SSCs). This understanding enables both the applicant and the NRC staff to properly evaluate the risk and safety-significance of SSCs during construction, ensuring that resources are strategically allocated to maintain facility safety and support informed regulatory oversight of construction-related changes.

Although the applicant is requesting an exemption from the financial requirements of 10 CFR 50.33(f) and 10 CFR Part 50, Appendix C, the requested exemption proposes to apply the 10 CFR Part 70 financial qualification requirements. The NRC staff's evaluation of the applicant's financial qualifications against the 10 CFR Part 70 requirements is provided in section 14.1 of this SE, in which the NRC staff concluded that LME had adequately demonstrated that it meets the 10 CFR Part 70 requirement. While the financial qualification criterion in 10 CFR Part 70 is different than the requirement in 10 CFR Part 50, the NRC staff has determined that the applicant meets the financial qualifications requirement in 10 CFR Part 70 and, that this demonstration, in combination with the applicant's use of the LMP approach, establishes that the applicant appears to be financially qualified to conduct the activities proposed in the CP application.

Given the NRC staff's established reactor oversight programs, the applicant's programs supporting the safe construction and operation of the facility, the proposed funding for construction of the LMGS provided in the financial support agreement, and the applicant's proposal to apply the 10 CFR Part 70 alternative financial qualification requirements for the CP application, the NRC staff has concluded that the issuance of the exemption would not have an adverse effect on public health and safety.

The Exemption is Consistent with the Common Defense and Security

The requested exemption pertains only to the financial qualification requirements for LMGS and does not affect any security requirements for LMGS. Therefore, the NRC staff has determined that the exemption request is consistent with the common defense and security.

Special Circumstances

The NRC staff evaluated whether special circumstances are present, as required in 10 CFR 50.12(a)(2). In its exemption request, LME asserts that the special circumstances required in 10 CFR 50.12(a)(2)(ii) are present, specifically that application of the financial qualification requirements in 10 CFR 50.33(f) and Appendix C, are not necessary to achieve the underlying purpose of the rule. As discussed above, the purpose of the financial qualification regulations for CPs in 10 CFR 50.33(f) and 10 CFR Part 50, Appendix C, is to prevent safety issues from occurring that might result from underfunding during construction.

In its exemption request and supplements, LME states that it is participating in the DOE's ARDP and that this meets the showing of special circumstances required in 10 CFR 50.12(a)(2)(vi). The NRC staff evaluated whether participation in DOE's ARDP is a special circumstance under 10 CFR 50.12(a)(2)(vi). As described in "Funding Opportunity Announcement-Advanced Reactor Demonstration Program," dated May 14, 2020, the ARDP is structured as a public-private partnership in which the DOE and industry jointly share the financial burden associated with the development and deployment of advanced nuclear reactors. Further, the NRC staff determined that the ARDP supports Executive Order 14300, "Ordering the Reform of the Nuclear Regulatory Commission," because section 2, "Policy," states, in part, that it is the policy of the United States to:

(a) Reestablish the United States as the global leader in nuclear energy;

(b) Facilitate increased deployment of new nuclear reactor technologies, such as Generation III+ and IV reactors, modular reactors, and microreactors, including by lowering regulatory and cost barriers to entry;

(c) Facilitate the expansion of American nuclear energy capacity from approximately 100 GW [gigawatt] in 2024 to 400 GW by 2050;

(d) Employ emerging technologies to safely accelerate the modeling, simulation, testing, and approval of new reactor designs.

ARDP funding is disbursed on a reimbursement basis, in incremental intervals, contingent upon the applicant's expenditure of funds. As the ARDP operates on a reimbursement model, the applicant must possess sufficient upfront capital to incur project costs prior to receiving ADRP reimbursement. Based on the ARDP funding and the applicant's financial qualifications evaluated by the NRC staff in section 14.2 of this SE, the NRC staff finds that safety lapses would not occur as a result of underfunding during the construction of the facility. Thus, the application of the financial qualification requirements in 10 CFR 50.33(f) and 10 CFR Part 50, Appendix C, is not necessary to achieve the underlying purpose of the regulation in this case, and special circumstances are present, in accordance with 10 CFR 50.12(a)(2)(ii).

The ARDP and the Executive Order are material circumstances that did not exist at the time the regulatory requirement for which the exemption is being sought, was issued. Because the ARDP aligns with the goals of the Executive Order, which establishes a national interest in the development of nuclear power alternatives such as the LMGS, the NRC staff finds that the objectives of the ARDP are clearly in the public interest. Furthermore, the NRC staff determined that the selection of LMGS by DOE as part of its ARDP funding supports the United States policy as described above, and therefore, the NRC staff determined that LME's requested exemption is in the public interest, in accordance with 10 CFR 50.12(a)(2)(vi).

Based on the above, the NRC staff finds that the application of the financial qualification requirements in 10 CFR 50.33(f) and 10 CFR Part 50, Appendix C, is not necessary to achieve the underlying purpose of the regulation in this case, and that special circumstances in accordance with 10 CFR 50.12(a)(2)(ii) and 10 CFR 50.12(a)(2)(vi), are present.

14.2.2 Conclusion

The NRC staff finds that an exemption from the financial qualifications requirements under 10 CFR 50.33(f), and 10 CFR Part 50, Appendix C, is acceptable because it is authorized by law, will not present undue risk to public health and safety, is consistent with common defense and security, and special circumstances are present.

14.3 Foreign Ownership, Control, or Domination

Section 50.38 of 10 CFR, "Ineligibility of Certain Applicants," states: "Any person who is a citizen, national, or agent of a foreign country, or any corporation, or other entity which the Commission knows or has reason to believe is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government, shall be ineligible to apply for and obtain a license." Section 2.4, "Foreign Ownership, or Control," of part 1 to the LMGS CP application describes the ownership and control of LME. This section of the CP application states that foreign entities do not possess the ability to vote or control decisions and cannot dominate LME.

14.3.1 Technical Evaluation

The NRC staff evaluated LME following the guidance provided in the "Final Standard Review Plan on Foreign Ownership, Control, or Domination" published in the *Federal Register* on September 28, 1999 (64 FR 52355), to determine whether an applicant is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government.

The LMGS CP application states that "LME is a U.S. company. […] LME is managed and controlled by TDCC [The Dow Chemical Company] via TDCC's wholly owned subsidiary GWNH [GWN Holding, LLC], an intermediate holding company that operates as a direct, wholly owned subsidiary of TDCC, both of which are also U.S. companies. In turn, TDCC is a wholly owned subsidiary of Dow Inc., a U.S. company that is publicly traded (NYSE: DOW) with no material foreign ownership of publicly traded shares." Additionally, LME provided the names, addresses, and citizenships of the TDCC and Dow, Inc. directors and principal officers, which are described in table I-3, and the names, addresses, and citizenships of the principal officers of GWNH, which are provided in table I-4.

The NRC staff conducted an independent analysis, including open-source research and verification of the information provided in the CP application, and found no evidence of foreign ownership, control, or domination (FOCD). Based on its review, the NRC staff does not know or have reason to believe that LME is owned, controlled, or dominated by a foreign interest.

14.3.2 Conclusion

In summary, based on the NRC staff's evaluation, the NRC staff concludes that the information provided by the applicant on FOCD is sufficient and that the application meets the requirements in 10 CFR 50.38.

14.4 Nuclear Insurance and Indemnity

LME did not provide nuclear insurance and indemnity information in its CP application. LME will be required to obtain insurance in accordance with 10 CFR 50.54(w) and to satisfy the financial protection and indemnity requirements in 10 CFR 140.11, "Amounts of financial protection for

certain reactors," prior to loading fuel. The NRC staff will review the sufficiency of LME's insurance and financial protection and indemnity provisions during its review of LME's expected operating license application.

15 EXEMPTIONS

Section 2.0, "Exemptions," of Part V, "Non-Applicabilities and Exemptions," (ML26261A427) of the Long Mott Generating Station (LMGS) construction permit (CP) application requested the NRC to grant exemptions from the following regulations upon the initiative of the Commission:

- 10 CFR 50.34(a)(4) as related to regulations for the emergency core cooling system (ECCS)

- Regulations relying on the definitions in 10 CFR 50.2 for safety-related (SR) structures, systems, and components (SSCs), reactor coolant pressure boundary, safe shutdown, and on the definition in Appendix A to 10 CFR Part 50 for important to safety (ITS) SSCs

Section 2.0 of Part V of the application also includes Long Mott Energy (LME) requests for exemption from the following regulations:

- 10 CFR 50.33(f)(1) and 10 CFR 50, Appendix C – financial qualifications
- 10 CFR 50.34(a)(1)(ii)(D) – analyses of postulated fission product releases
- 10 CFR 50.68 – criticality safety
- 10 CFR 26.4(f) – construction fitness for duty

15.1 Analysis and Evaluation of Emergency Core Cooling System (ECCS) Performance

15.1.1 Summary of Exemption

Section 2.1.1, "10 CFR 50.34(a)(4) – Analysis of Emergency Core Cooling System (ECCS) Performance," of Part V, "Non-Applicabilities and Exemptions," (ML26261A427) of the LMGS CP application identifies portions of 10 CFR 50.34(a)(4) as required for the preliminary safety analysis report (PSAR). The LMGS CP application requests that the Commission grant an exemption upon its initiative and states that, "The portions of 10 CFR 50.34(a)(4) requiring analysis and evaluation of [ECCS] performance in accordance with 10 CFR 50.46 is applicable to [light water reactors (LWRs)] and is not technically relevant to the LMGS non-LWR design."

15.1.2 Evaluation of Exemption

Authorized by Law

The NRC staff finds that granting of an exemption from the ECCS performance requirements in 10 CFR 50.34(a)(4) will not result in a violation of the Atomic Energy Act of 1954, as amended (hereafter referred to as the AEA), or the Energy Reorganization Act of 1979, as amended (hereafter referred to as the ERA). In addition, granting of an exemption from the ECCS performance requirements in 10 CFR 50.34(a)(4) will not result in a violation of the Commission's regulations because pursuant to 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 50.

Will not Present an Undue Risk to Public Health and Safety

The guidance in DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications-Roadmap," (ML23277A139), Appendix B, "Analysis of Applicability of NRC Regulations to Non-Light-Water Power Reactors," table 6, "Regulations Referencing Part 50 Regulations Limited to LWRs," states that, "The second sentence of [10 CFR 50.34(a)(4) and 10 CFR 50.34(b)(4)] requires a description of the analysis and evaluation of the ECCS cooling performance in accordance with 10 CFR 50.46, which is only applicable to LWRs." Accordingly, the NRC staff finds that granting the exemption from the ECCS performance requirements in 10 CFR 50.34(a)(4) will not present an undue risk to public health and safety because the requirements in 10 CFR 50.46 are not applicable to non-LWR designs and the portion of 10 CFR 50.34(a)(4) that references 10 CFR 50.46 is not technically relevant to the LMGS design.

Consistent with Common Defense and Security

The ECCS performance requirements in 10 CFR 50.34(a)(4) involve a design feature that is not part of the LMGS design and, therefore, does not affect design, function, or operation of any structures or plant equipment that is necessary to maintain a secure plant status. In addition, the requirements are not related to plant security or safeguards procedures or features. Therefore, as required by 10 CFR 50.12(a)(1), the NRC staff finds that the exemption is consistent with the common defense and security.

Special Circumstances are Present

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), are present whenever application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purpose of 10 CFR 50.34(a)(4) is to provide assurance of adequate core cooling under accident conditions. The LMGS design, as described in the PSAR, performs the required safety function of controlling heat removal without the use of an ECCS, which is only required for LWRs. Accordingly, the NRC staff finds that special circumstances are present under 10 CFR 50.12(a)(2)(ii) because the requirement is not necessary to achieve the underlying purpose of the rule.

15.1.3 Conclusion

Based upon the discussion above, the NRC staff determined that an exemption to the second sentence of 10 CFR 50.34(a)(4), requiring a description of the analysis and evaluation of the ECCS cooling performance in accordance with 10 CFR 50.46, is acceptable because it is authorized by law, will not present undue risk to public health and safety, is consistent with common defense and security, and special circumstances are present.

15.2 Regulations Relying on Definitions in 10 CFR 50.2 and Part 50, Appendix A

Section 2.1.2, "Regulatory Definitions," of Part V, "Non-Applicabilities and Exemptions," of the LMGS CP application requests that the Commission grant, on its own initiative, exemptions to the regulations that use 10 CFR 50.2 definitions for safety-related (SR) structures systems and components (SSCs), safety-related electrical equipment, reactor coolant pressure boundary (RCPB), safe shutdown, and important to safety (ITS). LME states that these are LWR-centric

definitions and proposes to use alternative definitions that are aligned with the LMGS design and the framework described in Nuclear Energy Institute (NEI) 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Licensing Basis Development," Revision 1 (ML19241A472) as endorsed by Regulatory Guide (RG) 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Application for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," Revision 0 (ML20091L698).

Because this exemption is being granted under the Commission's initiative, the information provided in section 2.1.2 of Part V of the application was used to inform the NRC staff's decisions but unless otherwise stated does not form the basis for the NRC staff's decisions.

The NRC staff identified instances where the definitions of SR SSCs, SR electrical equipment, RCPB, safe shutdown, and ITS appear in the regulations. The NRC staff evaluated the applicability of those regulations to the LMGS CP application and identified the exemptions warranted at the CP stage.

Safety-Related Structures, Systems, and Components

SR SSCs are defined for 10 CFR Part 26 in 10 CFR 26.5, and the term SR appears in §§ 26.3, 26.4, 26.403, 26.405, 26.406, 26.407, 26.409, and 26.419. These regulations are all applicable to LMGS' application.

In 10 CFR Part 50, the term is defined in 10 CFR 50.2 and appears without further definition in §§ 50.10, 50.55a, 50.69, Appendix B, Appendix R, and Appendix S. The same language used in the 10 CFR 50.2 definition of SR SSCs is also used in the 10 CFR 50.49(b)(1) definition of SR electrical equipment, the 10 CFR 50.65(b)(1) definition of SR SSCs within the scope of maintenance monitoring programs required by 10 CFR 50.65(a)(1), and in the 10 CFR Part 50 Appendix S definition of SSCs required to withstand the effects of the safe shutdown earthquake (SSE). Of these, 10 CFR 50.2, 50.10, 50.69, Appendix B, and Appendix S are applicable to LMGS' application; the reference to SR SSCs in 10 CFR 50.55a is not applicable to non-LWRs and 10 CFR 50.49 and 50.65 are not applicable to CPs.

The definition of SR SSCs for 10 CFR Part 73 is the same as the 10 CFR Part 50 definition, as stated in 10 CFR 73.2. In 10 CFR Part 73, the term SR appears in §§ 73.22, 73.54, and 73.77. Of these, only 10 CFR 73.22 is applicable to CPs.

Because the NRC staff identified several applicable regulations that use the term safety-related SSCs, the NRC staff initiated an exemption as detailed below.

Safety-Related Electrical Equipment

SR electrical equipment is defined in 10 CFR 50.49(b)(1)(i) with a definition that is identical to the definition of SR SSCs in 10 CFR 50.2. However, 10 CFR 50.49 is not applicable to CP applications. As such, the NRC staff did not consider exemptions to 10 CFR 50.49 in the scope of the NRC-initiated exemption discussed here.

Reactor Coolant Pressure Boundary

RCPB is defined in 10 CFR 50.2 in relation to “pressure-retaining components of boiling and pressurized water-cooled nuclear power reactors.” RCPB appears in regulations that are applicable to the LMGS application (a non-LWR applying for a CP) in the definition of SR SSCs and basic component in 10 CFR 50.2 and the definition of SSCs required to withstand the effects of the SSE in 10 CFR 50 Appendix S. Because the only applicable regulations that use RCPB use the term in a manner that is the same way it is used in the definition of SR SSCs in 10 CFR 50.2, the NRC staff is considering these regulations under the safety-related exemption and is not considering a separate exemption from the definition of RCPB.

Safe Shutdown

A general definition of the term safe shutdown is not provided in 10 CFR Part 50. 10 CFR 50.2 provides a definition for “safe shutdown (non-design basis accident (DBA)” specifically as it relates to station blackout conditions, but the requirement under 10 CFR 50.63 to consider station blackout is itself only applicable to LWRs. It is therefore acceptable for the applicant to identify an appropriate definition of safe shutdown consistent with applicable NRC policy and guidance to use where necessary in demonstrating compliance with the regulations. However, the NRC staff notes that the only instances of the term “safe shutdown” in the 10 CFR Part 50 regulations that are applicable to the LMGS CP application are in the context of the definition of safety-related equipment (or other terms that use the same definition, as discussed above) that are covered by the NRC-initiated exemption. Other instances are not applicable to new reactors, non-LWRs, or CP applicants. As such the NRC staff did not consider a separate exemption related to the definition of safe shutdown.

Important to Safety

The introduction to 10 CFR 50 Appendix A provides a definition of “important to safety” as “structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.” The NRC staff notes that 10 CFR 50.49(b) provides a more specific definition of “electrical equipment important to safety” as SR electrical equipment (following the same definition as in 10 CFR 50.2 for SR SSCs) plus certain non-SR electrical equipment. However, 10 CFR 50.49 is not applicable to CP applicants. As noted in DANU-ISG-2022-01 and Regulatory Guide (RG) 1.253, SSCs classified as SR and non-safety-related with special treatment (NSRST) under the NEI 18-04 process are considered important to safety. In section 2.1.2.4 of Part V of the application, LME described that “important to safety” SSCs for LMGS includes all SR and NSRST SSCs, consistent with the discussion in DANU-ISG-2022-01 and RG 1.253. As such, the NRC staff did not consider an exemption to apply an alternate definition of the term “important to safety” to be necessary for LMGS.

15.2.1 Summary of Exemption

LME uses the guidance in NEI 18-04, Revision 1, as endorsed by RG 1.233, Revision 0 (ML20091L620), to perform the safety classification of SSCs. Following this process, SSCs and associated functions may have different safety classifications than those based on the 10 CFR 26.5 and 10 CFR 50.2 definitions of safety-related SSCs. The NEI 18-04 approach to SSC classification is a risk-informed, performance-based, and technology-inclusive means of identifying safety-significant SSCs at the facility. Regulatory position C.2 of RG 1.233 states, in part, that use of the term “safety-related” in NEI 18-04 is not the same as the 10 CFR 50.2 definition, that SSCs included in the “safety-related” classification for non-LWRs may not be the

same as those considered safety-related for LWRs, and that applicants are expected to identify exceptions to and exemptions needed from NRC regulations. The NRC staff considers it appropriate for a non-LWR applicant to use the NEI 18-04 process to address whether an SSC should be considered safety-significant, based on the acceptability of the NEI 18-04 process as discussed in RG 1.233. Therefore, the NRC staff identified and concluded that granting an exemption from certain requirements related to the 10 CFR 26.5 and 10 CFR 50.2 definition of safety-related SSCs was warranted.

Though definitions themselves do not create requirements, the use of an alternate definition necessitates an exemption from the regulations where that definition is used.

The alternate definition of safety-related SSCs for that in 10 CFR 26.5 will be as follows for the applicant:

> …for the purposes of this part, those structures, systems, and components (SSCs) classified as safety-related in accordance with NEI 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Licensing Basis Development," Revision 1, as endorsed by Regulatory Guide 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," Revision 0.

The alternate definition of safety-related SSCs for that in 10 CFR 50.2 will be as follows for the applicant:

> Safety-related structures, systems, and components means those structures, systems and components classified as safety-related in accordance with NEI 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Licensing Basis Development," Revision 1, as endorsed by Regulatory Guide 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," Revision 0.

The NRC staff notes that 10 CFR Part 73 references the definitions from 10 CFR Part 50. The same alternate definition is referenced and evaluated below with respect to 10 CFR Part 73.

The alternate definition of "structures, systems, and components required to withstand the effects of the safe-shutdown earthquake ground motion or surface deformation" for that in 10 CFR 50 Appendix S will be as follows for the applicant:

> Structures, systems, and components required to withstand the effects of the safe-shutdown earthquake ground motion or surface deformation are those classified as safety-related in accordance with NEI 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Licensing Basis Development," Revision 1, as endorsed by Regulatory Guide 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," Revision

> 0, as well as those other SSCs required to perform a safety function during and after vibratory ground motion.

The NRC staff identified where the definition for "safety-related SSCs," is used in the regulations that apply the same language as those definitions, and regulations that refer to safety-related SSCs in 10 CFR parts 26, 50, and 73 that are relevant to a CP application. The following sections describe the regulations affected by the exemption:

10 CFR Part 26, "Fitness for Duty Programs"

10 CFR 26.3, "Scope," defines the scope of fitness for duty program requirements.

10 CFR 26.3(c)(4) specifies that the scope includes construction permit holder (under 10 CFR Part 50) and makes several references to SR SSCs.

10 CFR 26.4, "FFD program applicability to categories of individuals," provides requirements on the applicability of fitness for duty (FFD) programs to categories of individuals.

10 CFR 26.4(e)(2) refers to quality assurance, quality control, and quality verification activities related to SR SSCs, and 10 CFR 26.4(e)(5) refers to supervising or managing the construction of SR SSCs. 10 CFR 26.4(f) refers to constructing or directing the construction of SR SSCs.

10 CFR 26.5, "Definitions," defines safety-related SSCs as:

> …for the purposes of this part, those structures, systems, and components that are relied on to remain functional during and following design basis events to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in 10 CFR 50.34(a)(1).

10 CFR 26.403, "Written policy and procedures," provides requirements pertaining to FFD program policies and procedures. 10 CFR 26.403(b)(2) and (3) relate to the construction of SR SSCs.

10 CFR 26.405, "Drug and alcohol testing," provides requirements related to drug and alcohol testing for individuals involved in activities related to safety-related SSCs.

10 CFR 26.406, "Fitness monitoring," provides a requirement for monitoring the fitness for duty of individuals constructing or directing the construction of safety-related SSCs.

10 CFR 26.407, "Behavioral observations," provides a requirement for individuals involved in construction of safety-related SSCs to be subject to behavioral observation.

10 CFR 26.409, "Sanctions," requires licensees and other entities implementing an FFD program to establish sanctions for FFD policy violations that prohibit certain individuals from being involved in the construction of safety-related SSCs.

10 CFR 26.419, "Suitability and fitness evaluations," provides a requirement for licensees and other entities implementing fitness-for-duty programs to develop, implement, and maintain

procedures for evaluating whether to assign individuals to construct safety- and security-related SSCs.

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"

10 CFR 50.2 defines safety-related SSCs as follows:

Safety-related structures, systems and components mean those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary.

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition.

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

10 CFR 50.10, "License required; limited work authorization," paragraphs (a)(1)(i), (iii), and (iv) define construction activities in terms of safety-related SSCs, SSCs whose failure could prevent safety-related SSCs from fulfilling their safety-related function, and SSCs whose failure could cause a reactor scram or actuation of a safety-related SSC.

10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components," provides a voluntary alternative to compliance with a number of NRC regulations (identified in 10 CFR 50.69(b)(1)) for certain safety-related and non-safety-related SSCs that perform low safety significant functions.

10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," provides requirements related to quality assurance for "all activities affecting safety-related functions" of SSCs that "prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public."

10 CFR 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," III, "Definitions," provides the following definition:

> Structures, systems, and components required to withstand the effects of the safe shutdown earthquake ground motion or surface deformation are those necessary to assure:
>
> (1) The integrity of the reactor coolant pressure boundary;
>
> (2) The capability to shut down the reactor and maintain it in a safe-shutdown condition.
>
> (3) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of § 50.34(a)(1)."

Appendix S, section IV, “Application to Engineering Design,” also states the following relative to SSC response to the safe-shutdown earthquake:

(a)(1) Safe Shutdown Earthquake Ground Motion.

(ii) The nuclear power plant must be designed so that, if the Safe Shutdown Earthquake Ground Motion occurs, certain structures, systems, and components will remain functional and within applicable stress, strain, and deformation limits. In addition to seismic loads, applicable concurrent normal operating, functional, and accident-induced loads must be taken into account in the design of these safety-related structures, systems, and components. The design of the nuclear power plant must also take into account the possible effects of the Safe Shutdown Earthquake Ground Motion on the facility foundations by ground disruption, such as fissuring, lateral spreads, differential settlement, liquefaction, and landsliding, as required in § 100.23 of this chapter.
…
(iv) The evaluation must take into account soil-structure interaction effects and the expected duration of vibratory motion. It is permissible to design for strain limits in excess of yield strain in some of these safety-related structures, systems, and components during the Safe Shutdown Earthquake Ground Motion and under the postulated concurrent loads, provided the necessary safety functions are maintained.

10 CFR Part 73, “Physical protection of plants and materials”

10 CFR 73.22, “Protection of Safeguards Information: Specific requirements,” provides requirements related to the protection of Safeguards Information (SGI). 10 CFR 73.22(a)(1)(vii) specifically requires physical protection for documents that contain lists or locations of certain safety-related equipment.

15.2.2 Evaluation of Exemption

Exemption from 10 CFR Part 26
Pursuant to 10 CFR 26.9, “Specific exemptions,” the Commission may, on its own initiative, grant exemptions from the requirements of 10 CFR Part 26 as it determines are authorized by law, will not endanger life or property or the common defense and security, and are otherwise in the public interest.

Authorized by Law

The scope of SSCs considered to be SR is not defined in the Atomic Energy Act of 1954, as amended, or the Energy Reorganization Act of 1974, as amended. The identified exemption in this instance is authorized by law, because no other prohibition of law exists to preclude the activities which would be authorized by the exemption. Additionally, 10 CFR 26.9 allows the Commission to, on its own initiative, grant exemptions from the requirements of 10 CFR Part 26 if the Commission makes certain findings.

As such, the NRC staff has determined that granting an exemption from the requirement to apply the substantive regulations to safety-related SSCs meeting the definition in 10 CFR Part 26 for the purpose of complying with the above stated regulations in 10 CFR Part 26 and instead, use the alternate definition provided above, when identifying safety-related SSCs in the

sections of 10 CFR Part 26 identified above will not result in a violation of any Act or the Commission's regulations. Therefore, the exemption is authorized by law.

Will not Endanger Life or Property

As stated in RG 1.233, the NEI 18-04 definition of safety-related SSCs differs from that provided in the regulations, but "the methods described in NEI 18-04 constitute one acceptable means to… classify SSCs." RG 1.233 further states that the purpose of the safety classification aspect of NEI 18-04 is to ensure "applicants provide the necessary capabilities to perform mitigation functions and ensure the reliability of SSCs to prevent LBEs with more severe consequences" and "determine if special treatments beyond normal industrial practices are needed to ensure SSC performance of safety functions in the prevention and mitigation of LBEs." As discussed in NEI 18-04, these special treatments include provisions for the design, construction, installation, operation, and maintenance of SSCs that provide increased assurance beyond normal industrial practices that SSCs perform their design-basis functions.

Though the NEI 18-04 definition differs from the 10 CFR 26.5 definition of safety-related SSCs, it is consistent with the underlying purpose of identifying those SSCs that are subject to the strictest regulatory treatment to ensure they have adequate assurance that they can perform their design-basis functions. A more comprehensive evaluation of the NEI 18-04 definition of safety-related and its consistency with public health and safety is provided below with respect to the definition provided in 10 CFR 50.2 (which is identical to that provided in 10 CFR 26.5). The above 10 CFR Part 26 regulations to which this exemption would apply are related to the FFD program. This exemption would ensure that the FFD program applies to those SSCs that are subject to the strictest regulatory treatment, which is consistent with the underlying intent of the definition of safety-related SSCs in 10 CFR Part 26.

As discussed in PSAR section 5.1.1 and evaluated in section 5.1.1 of this SE, LME deviates from the NRC staff position C.2 in RG 1.233 for the safety classification of SSCs providing a load path for SR SSCs, which should be classified in a manner consistent with the higher-level function they are supporting (i.e., SR) but are classified as risk-significant NSRST by LME with additional treatments applied to ensure they can perform their safety-significant function. As discussed in section 5.1.1 of this SE, the foundations of these structures are within the scope of the FFD program because the definition of constructing or construction activities in 10 CFR Part 26 explicitly includes installation of the foundations, including concrete, for safety-related SSCs. As such, the proposed exemption will not endanger life or property because safety-related SSCs will continue to be defined using a process the NRC staff has determined to be acceptable and will remain subject to appropriate fitness-for-duty programs to ensure they are constructed appropriately.

Will not Endanger the Common Defense and Security

The exemption does not affect requirements in 10 CFR Part 26 related to nuclear power reactor security; therefore, this exemption will not endanger the common defense and security. Requirements in 10 CFR Part 73 that are affected by the exemption, and which could relate to the common defense and security are evaluated below.

In the Public Interest

The FFD program ensures that personnel are fit to safely and competently perform their duties. This exemption is in the public interest because it would ensure that the FFD program applies to the most safety-significant SSCs.

Exemption from 10 CFR Part 50

Pursuant to 10 CFR 50.12, the NRC may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 50 when: (1) the exemptions are authorized by law, will not present an undue risk to the public health or safety, and are consistent with the common defense and security; and (2) when special circumstances are present. Under 10 CFR 50.12(a)(2), special circumstances include, among other things, when application of the specific regulation in the particular circumstances would not serve, or is not necessary to achieve, the underlying purpose of the rule.

Authorized by Law

The scope of SSCs considered to be SR is not defined in the Atomic Energy Act of 1954, as amended, or the Energy Reorganization Act of 1974, as amended. The identified exemption in this instance is authorized by law, because no other prohibition of law exists to preclude the activities which would be authorized by the exemption. Additionally, 10 CFR 50.12 allows the Commission to, on its own initiative, grant exemptions from the requirements of 10 CFR Part 50 if the Commission makes certain findings.

As such, the NRC staff has determined that granting an exemption from the requirement to apply the substantive regulations to safety-related SSCs meeting the definition in 10 CFR Part 50 for the purpose of complying with the above stated regulations in 10 CFR Part 50 and instead, use the alternate definition provided above, when identifying safety-related SSCs in the sections of 10 CFR Part 50 identified above will not result in a violation of any Act or the Commission's regulations. Therefore, the exemption is authorized by law.

Will not Present an Undue Risk to Public Health and Safety

Though the NEI 18-04 definition differs from the 10 CFR 50.2 definition of SR SSCs, it is consistent with the underlying purpose of identifying SSCs that are subject to the strictest regulatory treatment to ensure they have adequate assurance that they can perform their design-basis functions. The 10 CFR 50.2 definition of "safety-related" contains three elements which, as noted above, relate to the maintaining the integrity of the RCPB, achieving and maintaining safe shutdown, and preventing and mitigating radiological consequences of accidents. With respect to the first and second elements, the NEI 18-04 process includes the evaluation and classification of those SSCs necessary to support three fundamental safety functions (FSFs), including control of heat generation, control of heat removal, and radionuclide retention. The NEI 18-04 process evaluates the SSCs necessary to achieve these FSFs and classifies them as SR if they are necessary to maintain the frequencies and consequences of DBEs and certain BDBEs below a target derived from NRC regulations, including 10 CFR 50.34 and 10 CFR Part 20. With respect to the third element, the NEI 18-04 process explicitly includes an evaluation of DBAs to ensure that their radiological consequences are maintained below the 10 CFR 50.34(a)(1)(ii)(2) dose criteria; those SSCs identified as necessary to maintain consequences below the acceptance criterion are classified as SR. Based on this, the NRC staff

determined that the NEI 18-04 process includes appropriate consideration of the key elements of the 10 CFR 50.2 definition of safety-related, particularly including the consequences of accidents. As such the NRC staff concluded that replacing this portion of the 10 CFR 50.2 definition with the NEI 18-04 definition of safety-related, as delineated above for the applicant, would not present an undue risk to public health and safety.

The purpose of 10 CFR 50.10 is to define construction (10 CFR 50.10(a)); to provide a requirement for a license for production and utilization facilities (10 CFR 50.10(b)); to specify that construction cannot begin until a construction permit, combined license, early site permit, or limited work authorization (LWA) is issued (10 CFR 50.10(c)); to provide requirements concerning the request for, issuance of, and effect of LWAs (10 CFR 50.10(d) through (f)); and to provide requirements for implementation of a redress plan (10 CFR 50.10(g)). The current action would provide LME with a CP for LMGS, with the finding that construction would not present an undue risk to public health and safety pursuant to 10 CFR 50.35(a)(4)(ii) and enabling construction activities to commence. As such, this exemption does not present an undue risk to public health and safety as it relates to 10 CFR 50.10.

10 CFR 50.69 is a voluntary alternative to compliance with several regulations and is not identified in the LMGS application. Because of this, the NRC staff did not consider it in the present evaluation.

Regarding Appendix B to 10 CFR Part 50, using an alternate definition of SR SSCs does not affect the need for quality assurance (QA) for SSCs that “prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.” Specifically, SR SSCs identified under the NEI 18-04 process include those SSCs that serve to prevent or mitigate the consequences of design basis accidents to within the 10 CFR 50.34 acceptance criteria, and other SSCs as necessary to maintain consequences from design basis events within more stringent acceptance criteria (depending on frequency) and beyond-design basis events that have the potential to exceed the 10 CFR 50.34 acceptance criteria below an acceptable frequency. The NRC staff also notes that the QA program, which is described in PSAR section 8.1 and was found acceptable in section 8.1 of this SE, is applied to both SR and non-SR SSCs as described in PSAR section 8.1. Under this program, QA for SR SSCs meets the requirements of 10 CFR 50 Appendix B. As described in Part III of the applicant’s approved quality assurance program description (QAPD), which is incorporated by reference into the PSAR as stated in PSAR section 8.1, non-SR SSCs receive a graded approach to QA that applies controls to those characteristics or critical attributes that render the SSC a significant contributor to plant safety. Based on these considerations, the NRC staff determined that implementation of the modified definition of SR SSCs would not present an undue risk to public health and safety as it relates to 10 CFR 50 Appendix B.

Regarding Appendix S to 10 CFR 50, the modification to the definition of SR SSCs would provide an alternate scope of SSCs required to withstand the effects of the safe shutdown earthquake (SSE) ground motion or surface deformation to SR SSCs identified following NEI 18-04, as well as those other SSCs required to perform a safety function during and after vibratory ground motion based on their risk- and safety-significance. The changes from “safety-related” to the alternate language of “certain” SSCs in the context of the requirements for section IV of Appendix S are needed to conform to this scope. The NRC staff considers it important for certain non-SR SSCs to be included in the scope of Appendix S because, under the risk-informed and performance-based framework of NEI 18-04, NSRST SSCs may need to perform functions following an SSE that contribute to plant safety and all SSCs should be

assessed for potential seismic interactions. LME's process for seismic design and classification is described in PSAR section 3.7. Under that process, SSCs are categorized as seismic category I (SC-I), seismic category IIA (SC-IIA), seismic category IIB (SC-IIB), and seismic category III. As discussed in section 3.7 of this SE, the criteria proposed by LME for assigning the initial seismic categories to SSCs considers the safety function of the SSCs, their seismic risk significance, their contribution to adequate DID in response to a seismic event, **[[ ]]**, and the potential for seismic interaction. In the PSAR, all SSCs that perform a required safety function – that is, all SR SSCs – are categorized as SC-I. SC-I SSCs are designed to withstand the effects of the SSE. Some SC-IIA SSCs are designed to **[[ ]**; other SC-IIA SSCs are designed to **[[ ]]**. SC-IIB is applied to SSCs that may seismically challenge SC-I and SC-IIA SSCs. SC-III is applied to other SSCs. The NRC staff evaluated this process and found it to be acceptable in SE section 3.7. Based on this, the NRC staff determined that the modifications to use an alternate definition of SSCs required to withstand the effects of the safe-shutdown earthquake ground motion or surface deformation to that in 10 CFR 50 Appendix S do not present an undue risk to public health and safety.

For the reasons discussed above relative to each affected regulation in 10 CFR Part 50, the NRC staff determined that the proposed exemption will not present an undue risk to public health and safety.

Will be Consistent with Common Defense and Security

The exemption does not affect requirements in 10 CFR Part 50 related to nuclear power reactor security; therefore, this exemption will not endanger the common defense and security. Requirements in 10 CFR Part 73 that are affected by the exemption, and which could relate to the common defense and security are evaluated below.

Special Circumstances are Present

Special circumstances exist pursuant to 10 CFR 50.12(a)(2)(ii) because application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule. Applying the original definition of SR SSCs in 10 CFR 50.2, 50.10, Appendix B, and Appendix S is unnecessary to achieve the underlying purpose of these rules because SR SSCs will continue to be defined using a process found acceptable by the NRC staff that appropriately ensures safety, as discussed above. The requirements for SR SSCs in each section will still apply, and the modifications to use an alternate scope of 10 CFR Appendix S appropriately address non-SR SSCs that perform or could affect the ability of other SSCs to perform safety-significant functions.

Exemption from 10 CFR Part 73

Pursuant to 10 CFR 73.5, "Specific exemptions," the Commission may, upon application of any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 73 as it determines are authorized by law, will not endanger life or property or the common defense and security, and are otherwise in the public interest.

Authorized by Law

The scope of SSCs considered to be SR is not defined in the Atomic Energy Act of 1954, as amended, or the Energy Reorganization Act of 1974, as amended. The identified exemption in this instance is authorized by law, because no other prohibition of law exists to preclude the activities which would be authorized by the exemption. Additionally, 10 CFR 73.5 allows the Commission to, on its own initiative, grant exemptions from the requirements of 10 CFR Part 73 if the Commission makes certain findings.

As such, the NRC staff has determined that granting an exemption from the requirement to apply the substantive regulations to safety-related SSCs meeting the definition in 10 CFR Part 50 for the purpose of complying with the above stated regulation in 10 CFR Part 73 and instead, use the alternate definition discussed above, when identifying safety-related SSCs in the section of 10 CFR Part 73 identified above will not result in a violation of any Act or the Commission's regulations. Therefore, the exemption is authorized by law.

Will not Endanger Life or Property or the Common Defense and Security

10 CFR 73.22(a)(1)(vii) requires the physical protection of documents and other matters that contain lists or locations of certain safety-related physical equipment explicitly identified in the documents or other matter as vital for purposes of physical protection, as contained in security plans, contingency measures, or plant specific safeguards analyses. Though SSCs identified as NSRST using the NEI 18-04 process may perform safety- or risk-significant functions, SSCs identified as SR using the NEI 18-04 process are those most important to ensure the safety of the facility. These SSCs would continue to be included in the scope of protected information under 10 CFR 73.22. As such, the NRC staff concluded that the proposed exemption will not endanger life or property or the common defense and security as it relates to 10 CFR 73.22.

In the Public Interest

Safeguards information concerns the physical protection of facilities, spent fuel shipments, and other radioactive material. This exemption is in the public interest because it would ensure that relevant information is protected as safeguards information.

15.2.3 Conclusion

As discussed above, the NRC staff determined that in accordance with 10 CFR 26.9, the exemption is authorized by law, will not endanger life or property, will not endanger the common defense and security, and is otherwise in the public interest. The NRC staff additionally determined that in accordance with 10 CFR 50.12, the exemptions from Part 50 are authorized by law, would not present an undue risk to public health and safety, are consistent with the common defense and security, and special circumstances are present pursuant to 10 CFR 50.12(a)(2)(ii). Finally, the NRC staff also determined that, in accordance with 10 CFR 73.5, the exemption is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest.

15.3 Financial Qualifications

In section 2.2.2, "10 CFR 50.33(f)(1) and 10 CFR 50, Appendix C – Financial Qualifications," of Part V, "Non-Applicabilities and Exemptions," of the LMGS CP application, LME requests an exemption from elements of 10 CFR 50.33(f)(1) and 10 CFR 50, Appendix C. This exemption request is evaluated in chapter 14 of this SE.

15.4 Analysis of Postulated Fission Product Release

15.4.1 Summary of Exemption

In section 2.2.2, "10 CFR 50.34(a)(1)(ii)(D) – Analysis of Postulated Fission Product Release," of Part V, "Non-Applicabilities and Exemptions," (ML26261A427) of the LMGS CP application, LME requests an exemption from elements of 10 CFR 50.34(a)(1)(ii)(D). The request states that the existing regulation prescribes analysis methodology and assumptions which are not technically relevant to the Xe-100 design and are not consistent with the licensing modernization project (LMP)-based methodology described in NEI 18-04. Specifically, 10 CFR 50.34(a)(1)(ii)(D) requires the applicant to analyze an assumed fission product release from the core into the containment as a result of a "major accident," which per Footnote 6 of Paragraph (a)(1)(ii)(D) is assumed to be an accident that results in a substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.
LME proposes to use the DBA dose consequence results from its LMP-based analysis approach per NEI 18-04 to establish the acceptability of the LMGS exclusion area boundary (EAB) and low population zone (LPZ). The DBA analysis under an LMP-based approach is a deterministic, conservative analysis that is analogous to the DBA analyses performed for new LWRs and operating reactors. The proposed approach is similar to Option 1 included in Staff Position C.3.b(1) of RG 1.253 as an acceptable means for a non-LWR applicant using the LMP-based methodology to address the requirements of 10 CFR 50.34(a)(1)(ii)(D).

15.4.2 Evaluation of Exemption

Authorized by Law

The NRC staff has determined that granting the applicant's proposed exemption will not result in a violation of the Atomic Energy Act of 1954, as amended (hereafter referred to as the AEA), or the Energy Reorganization Action of 1979, as amended (hereafter referred to as the ERA). In addition, granting of an exemption from the analysis of postulated fission product release in 10 CFR 50.34(a)(1)(ii)(D) will not result in a violation of the Commission's regulations because pursuant to 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 50.

Will not Present an Undue Risk to Public Health and Safety

In Staff Position C.3.b of RG 1.253, the NRC staff states under Option 1 that the use of DBA dose consequence results from an LMP-based approach to establish the acceptability of the EAB and LPZ is one of the two possible options for addressing total calculated radiological dose at EAB, LPZ, and the control room (if required) to demonstrate that the facility meets the requirements of 10 CFR 50.34(a)(1)(ii)(D) or 10 CFR 52.79(a)(1)(vi) and the PDC for the control room. The section of the RG provides rationale that the DBA analysis under an LMP-based

approach is a deterministic, conservative analysis that is analogous to the DBA analysis performed for new LWRs and operating reactors. This section of the RG further outlines the nature of the conservative evaluation that is to be performed and states that the result is acceptable if the total calculated radiological consequences for the postulated fission product release meet the reference values for public dose, given in 10 CFR 50.34(a)(1)(ii)(D) and 10 CFR 52.79(a)(1)(vi).

LME states in the exemption request in section 2.2.2 of Part V, “Non-Applicabilities and Exemptions,” (ML26261A427) of the LMGS CP application that the analysis performed in PSAR section 3.2 and the associated topical reports demonstrate that applying the NRC RG 1.253, Staff Position C.3.b(1), Option 1 methodology to the analysis required by 10 CFR 50.34(a)(1)(ii)(D) has no adverse impact on public health and safety.

Based on the review, the NRC staff determined that granting the exemption will not present an undue risk to public health and safety because the analysis methods, assumptions and the associated results presented in PSAR section 3.2 show that none of the DBAs result in EAB pr LPZ dose greater than 25 rem (0.25 sievert (Sv)) consistent with the limits in 10 CFR 50.34(a)(1)(ii)(D) and 10 CFR 52.79(a)(1)(vi).

Consistent with Common Defense and Security

The proposed exemption does not affect design, function, or operation of any structures or plant equipment that is necessary to maintain a secure plant status. In addition, the proposed exemption has no impact on plant security or safeguards procedures. Therefore, as required by 10 CFR 50.12(a)(1), the NRC staff finds that the common defense and security is not impacted by this exemption.

Special Circumstances are Present

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), are present whenever application of the regulation in the circumstances would not serve the underlying purpose of the rule or is not necessary to achieve it. The purpose of 10 CFR 50.34(a)(1)(ii)(D) is protection of the health and safety of the public by providing assurance that the total dose values remain below the values deemed acceptable by regulation. The exemption request states that the fission product release analysis includes an LWR-centric methodology required under 10 CFR 50.34(a)(1)(ii)(D) that is not practicable to apply to LMGS design, which includes a functional containment concept and has a very different core damage and fission product release modes.

Accordingly, the NRC staff finds that special circumstances are present under 10 CFR 50.12(a)(2)(ii) since the regulation contains language that is specific to LWR designs is not practicable to apply to advanced reactor design with a functional containment concept and core damage pathways.

15.4.3 Conclusion

The NRC staff finds that an exemption from elements of 10 CFR 50.34(a)(1)(ii)(D) which prescribe that an applicant analyze an assumed fission product release from the core into the containment as a result of a “major accident” is acceptable because it is authorized by law, will not present undue risk to public health and safety, is consistent with common defense and security, and special circumstances are present.

15.5 Criticality Safety

15.5.1 Summary of Exemption

Section 2.2.3, "10 CFR 50.68 – Criticality Safety," of Part V, "Non-Applicabilities and Exemptions," (ML26261A427) of the LMGS CP application identifies portions of 10 CFR 50.68 as required for the PSAR. Under 10 CFR 50.68(a), construction permit holders must comply with either 10 CFR 70.24 or 10 CFR 50.68(b). Those choosing to comply with 10 CFR 50.68(b) may do so in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24. LME has requested an exemption that provides alternative requirements for six of the eight requirements under 10 CFR 50.68(b) as shown in table 15.5-1 below.

Table 15.5-1: 10 CFR 50.68(b) Proposed Alternative Requirements

10 CFR 50.68(b) Requirement	**Proposed Alternative Requirement**
(b)(1) Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.	(b)(1) Plant procedures shall prohibit the handling and storage at any one time of more fuel pebbles that have been determined to be safely subcritical under the most adverse moderation conditions feasible for unborated water.
(b)(2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.	(b)(2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of fresh fuel in fresh fuel storage areas shall be calculated assuming fuel pebbles in the area are at maximum fuel pebble reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.
(b)(3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used.	(b)(3) If optimum moderation of fresh fuel in fresh fuel storage areas occurs when the fuel pebbles in the area are assumed to be at maximum fuel pebble reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used.

(b)(4) If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.	(b)(4) The k-effective of spent fuel in areas containing or potentially containing spent fuel pebbles at the maximum fuel pebble reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.
(b)(7) The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight.	(b)(7) The maximum nominal U-235 enrichment of the fresh TRISO fuel is limited to less than twenty (20) percent by weight.
(b)(8) The FSAR is amended no later than the next update which § 50.71(e) of this part requires, indicating that the licensee has chosen to comply with § 50.68(b).	(b)(8) The FSAR is amended no later than the next update which § 50.71(e) of this part requires, indicating that the licensee has chosen to comply with § 50.68(b) to the extent practicable and as described in an approved exemption under 10 CFR 50.12, in lieu of maintaining a monitoring system as described in 10 CFR 70.24.

15.5.2 Evaluation of Exemption

Authorized by Law

The NRC staff has determined that granting of the applicant's proposed exemption will not result in a violation of the Atomic Energy Act of 1954, as amended (hereafter referred to as the AEA), or the Energy Reorganization Action of 1979, as amended (hereafter referred to as the ERA). In addition, granting of an exemption from the analysis of postulated fission product release in 10 CFR 50.68 will not result in a violation of the Commission's regulations because pursuant to 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 50.

Will not Present an Undue Risk to Public Health and Safety

Because changes being proposed to the other portions of 10 CFR 50.68 under this exemption do not change the level of criticality safety assessment required to be performed, the applicant must, for fuel up to the requested enrichment of less than 20 percent by weight: maintain appropriate fuel handling and storage procedures in accordance with revised 10 CFR 50.68(b)(1); demonstrate acceptable margin to criticality limits in accordance with revised 10 CFR 50.68(b)(2) through (4); maintain the quantity of special nuclear material (SNM) other than fuel stored on site to less than the quantity necessary for a critical mass; and provide radiation monitors in storage and handling areas. The NRC staff notes that a change in enrichment does

not affect the requirement for procedures under 10 CFR 50.68(b)(1), the requirement to maintain SNM other than fuel below a critical mass under 10 CFR 50.68(b)(5), or the requirement for radiation monitors under 10 CFR 50.68(b)(6). The NRC staff's evaluation therefore focuses on the equipment used to store fresh and spent fuel and the analysis used to demonstrate it remains subcritical.

Descriptions of the preliminary criticality safety analyses for areas of fresh fuel storage and spent fuel storage are provided in PSAR section 3.9 and include descriptions of the physical layout of fissile material, key assumptions used, and the resulting controls relied on to prevent criticality events. These analyses include criticality safety calculations for optimal moderation both inside and outside of the storage canisters. The NRC staff's evaluation of these analyses is provided in section 3.9 of this SE.

Most of the changes requested under this exemption are related to the change in fuel type from fuel assemblies to fuel pebbles. One primary difference between fuel assemblies and fuel pebbles is that pebbles can flow to unintended locations and collect if the canister or system holding them fails. NRC staff evaluated the impact of this on the exemption request and determined that the impacts of this can be effectively addressed through considering credible geometry upset conditions in the criticality safety evaluation. The NRC staff found that the preliminary analysis did include some consideration of credible geometry upset conditions and LME confirmed in audit that these geometry considerations are part of the method used for criticality safety evaluations.

The statements of consideration for 10 CFR 50.68 (63 FR 63127) addressed a comment that a limit on enrichment offered no "direct safety benefit" and should not have been included in the rule. The Commission disagreed, stating that the 5 percent limit was based on "possible criticality concerns even in a dry environment" as well as concerns related to the environmental review required by the National Environmental Policy Act (NEPA). Because the criticality analyses discussed above explicitly consider enrichment up to the proposed limit of less than 20 weight percent and model various configurations for the SSCs used to store fresh and spent fuel, including both dry and wet conditions, the NRC staff determined that the Commission's concern on inadvertent criticality for higher enrichments has been addressed. The NRC staff also confirmed in audit that while the exemption uses the term "maximum nominal" when referring to the revised limit of 20 percent, LME views this as a maximum, including consideration of uncertainty, consistent with how the limit of 5 percent has historically been interpreted.

Consistent with Common Defense and Security

The proposed exemption does not affect design, function, or operation of any structures or plant equipment that is necessary to maintain a secure plant status. In addition, the proposed exemption has no impact on plant security or safeguards procedures. Therefore, as required by 10 CFR 50.12(a)(1), the NRC staff finds that the common defense and security is not impacted by this exemption.

Special Circumstances are Present

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), are present whenever application of the regulation in the circumstances would not serve the underlying purpose of the rule or is not necessary to achieve it. The purpose of 10 CFR 50.68 is protection of the health

and safety of the public by providing assurance that criticality is detected or sufficiently prevented. The exemption request states that alternative requirements are appropriate for design-specific aspects that differ from the LWR-centric language in the regulation and meet the underlying purpose of the regulation. The application states that changes are made to address the use of fuel pebbles versus assemblies and the use of fresh fuel maximum nominal U-235 enrichment of less than 20 weight percent.

Accordingly, the NRC staff finds that special circumstances are present under 10 CFR 50.12(a)(2)(ii) since the regulation contains language that is specific to LWR designs and the alternate requirements meet the underlying purpose of the regulation.

15.5.3 Conclusion

The NRC staff finds that an exemption from elements of 10 CFR 50.68, which prescribes that an applicant shall either comply with the 10 CFR 70.24 requirements for maintaining a monitoring system capable of detecting a criticality or comply with the requirements in 50.68(b), is acceptable because it is authorized by law, will not present undue risk to public health and safety, is consistent with common defense and security, and special circumstances are present.

15.6 Construction Fitness for Duty (FFD)

15.6.1 Summary of Exemption

Section 2.2.4, "10 CFR 26.4(f) – Construction Fitness-for-Duty," of Part V, "Non-Applicabilities and Exemptions," (ML26261A427) of the LMGS CP application requests an exemption in accordance with the provisions of 10 CFR 26.9, "Specific Exemptions," to the 10 CFR 26.4(f) requirement that individuals who construct or direct the construction of safety- or security related structures, systems, and components (SSCs)[1] be subject to an FFD program meeting the requirements under Subpart K, "FFD Program for Construction" of 10 CFR Part 26, unless those individuals are subject to an FFD program meeting the requirements under Subparts A through H, N, and O of 10 CFR Part 26. The LMGS construction FFD program that would apply to 10 CFR 26.4(f) workers is described under PSAR section 11.6.

The NRC staff noted the following, regarding the exemption request:

- Section 2.2.4.3 states that a partial exemption to 10 CFR 26.4(f) would, "exclude escorted visitors who construct or direct the construction of safety- or security-related SSCs for 30 days or less in a 60-day period from the requirement to be subject to 10 CFR 26, Subpart K, FFD requirements." Section 2.2.4.5 further states that "prior to exceeding this administrative limit [30 days or less in a 60-day period] the construction worker would be removed from the escort program and subjected to the 10 CFR 26 FFD program."

- Section 2.2.4.4 states that the "[c]onstruction FFD and access control administrative procedures will be…consistent with the applicable guidance of NEI 06-06 ["Fitness for Duty Program Guidance for New Nuclear Power Plant Construction Sites" (Revision 6, ML13093A340)"] and NEI 09-01 ["Security Measures During New Reactor Construction" (Revision 0)], with control of escorted visitors and escort requirements meeting the intent

[1] The exemption from the definition of safety-related for 10 CFR Part 26 is discussed in section 15.2 of this SE.

of the operating plant requirements for visitors and escorts in 10 CFR 73.55(g)(7) and 10 CFR 73.55(g)(8), respectively." PSAR section 11.6.2, "table 11.6-1, Exceptions to NEI 06-06 to Implement Partial 10 CFR 26.4(f) Exemption," includes detail on the procedures LMGS will use to implement the escorting program. These procedural descriptions align with the requirements in 10 CFR 73.55(g)(7) and (8). Specifically, under section "5.2, Procedures," in table 11.6-1, the content that appears under "Implement procedures for processing, escorting, and controlling Construction Visitors" aligns with 10 CFR 73.55(g)(7) and the content that appears under "Implement procedures for training and controlling Construction Escorts" aligns with 10 CFR 73.55(g)(8).

- Section 2.2.4.3 states that "[t]o implement the exemption from 10 CFR 26.4(f)…, an exception is identified to the conformance to Regulatory Guide 5.84 to allow escorted visitors access to the construction site for 30 days in a 60-day period to construct or direct construction of safety- and security related SSC." PSAR section 11.6.2, "Table 11.6-1, Exceptions to NEI 06-06 to Implement Partial 10 CFR 26.4(f) Exemption," identifies the affected sections in NEI 06-06 (Revision 6) and describes each exception. The identified sections in NEI 06-06 (Revision 6) with exceptions include: "2.1, Construction", "3.1.2, Persons Constructing or Directing the Construction of Safety- or Security-Related SSCs", "4. Definitions", and "5.2, Procedures."

- Section 2.2.4.4 states that the "construction workers not subject to the 10 CFR 26 Subpart K FFD program will be escorted by and under the direct observation of individuals who are subject to the FFD program under 10 CFR 26, Subpart A through H, N, and O (see PSAR section 11.6.2), trained, and empowered to ensure that individuals performing activities that would otherwise subject them to 10 CFR 26.4(f): (1) follow LMGS-established work processes and procedures; and (2) do not demonstrate physical and mental impairment that could adversely affect their ability to safely and competently perform their duties."

- Section 2.2.4.5 states that "escorts will be trained, in part, to: (1) stop work if the escort identifies conditions that adversely affect a safety- or security-related SSC; (2) immediately communicate with security if a problem arises; (3) maintain direct observation and control of their construction workers under escort; (4) observe workers for unsafe and improper actions; and (5) employ behavioral observation techniques to detect aberrant behavior."

- Section 2.2.4.5 states that escorts will be "trained and empowered to ensure that individuals performing 10 CFR 26.4(f) duties and responsibilities are acting in accordance with LMGS-established work processes and do not demonstrate characteristics of an insider threat. Such a threat could be an individual who possesses both intent and access to sensitive information or SSCs, to aid or cause an attack, condition, or theft inimical to public health and safety or the common defense and security."

- Section 2.2.4.5 states that "[r]equiring all individuals who perform 10 CFR 26.4(f) duties and responsibilities to be subject to a 10 CFR 26, Subpart K, FFD program prior to the LMGS transition to an operational FFD program would have significant adverse construction cost and schedule impacts with no offsetting benefit to public and

environmental health and safety. The exemption would result in a schedule benefit because individuals would be immediately available to perform emergent work activities directly related to construction critical path, which would help avoid construction schedule delays and the associated increased costs associated with those delays. Furthermore, there would be a reduction in burden because the exemption would increase flexibility in the manner in which workers can be brought onsite during emergent construction activities."

- Section 2.2.4.1 states that "[t]he requested exemption is substantially equivalent to the exemption granted by the Commission for the Vogtle Electric Generating Plants Units 3 and 4 in June 2019" [ML19108AA242 and ML1918A244].

15.6.2 Evaluation of Exemption

Authorized by Law

The NRC staff has determined that granting the applicant's proposed exemption will not result in a violation of the AEA, as amended, or the ERA, as amended. In addition, granting of the applicant's proposed exemptions will not result in a violation of the Commission's regulations because pursuant to 10 CFR 26.9, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 26.

Will not Endanger Life or Property or the Common Defense and Security

The exemption enables the escorting of temporary construction workers performing duties under 10 CFR 26.4(f). The exemption does not change any SSC, any existing natural or man-made geological/biological feature, or other material property on the construction site, nor does the exemption introduce any new industrial, chemical, or radiological hazard that would present itself as a latent or apparent adverse effect on life or property. The exemption also does not request any relaxation in work controls, processes, quality assurance, quality verification, or the training, qualification, and personal injury protections afforded to individuals constructing or directing the construction of safety- or security-related SSCs.

Otherwise in the Public Interest

The NRC staff finds that it is in the public's interest that LME be allowed to escort certain individuals who direct or perform 10 CFR 26.4(f) duties and responsibilities because the proposed exemption has a direct nexus to improve flexibility in construction scheduling, work planning, and conduct of construction activities. These improvements occur because the exemption would enable LME to implement process and procedural changes in a manner that leverages immediately available construction workers for short duration construction activities without a reduction in the assurance of public health and safety or the common defense and security because of the LME-proposed administrative controls for its escort program.

15.6.3 Conclusion

The NRC staff finds that the partial exemption to 10 CFR 26.4(f), as described in section 2.2.4 of Part V of the LMGS CP application (ML26261A427) is acceptable because it is authorized by law, it will not endanger life or property or the common defense and security, and it is otherwise in the public interest.

APPENDIX A – CONSTRUCTION PERMIT CONDITIONS

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that the Long Mott Generating Station construction permit (CP) should be conditioned to require that the applicant: (1) perform geologic mapping of the excavations for the safety-related engineered structures and notify the NRC staff when this information is available for examination, and (2) notify the NRC staff when site grading and engineered flooding protection design is available for examination. Therefore, the NRC staff recommends that, should the permit be granted, the CP include the conditions identified in this appendix. Additional details on the basis for each of these conditions are provided in the SE sections identified below.

Proposed Permit Condition	SE Section	Description
1	2.4, "Hydrology"	LME shall notify the Director of the Office of Advanced Reactors, or the Director's designee, as specified in 10 CFR 50.4, when the site grading design and the passive engineered flooding protection design that may impact site grading are available for examination by the NRC staff. This is expected to occur prior to the placement of safety-significant concrete for structure foundations. The design should consider the primary site flood hazard causing mechanisms described in PSAR section 2.4. The design should also demonstrate that safety-related structures, systems, and components (SSCs) and the subset of non-safety-related with special treatment SSCs that must remain functional during and after the design basis flood event are protected. Aspects of site grading and engineered flooding protection features that may impact the results of previous flooding analyses (i.e., design inputs and assumptions) should be addressed such that the design and analyses are consistent.
2	2.5.1.2, "Proposed Permit Condition"	Long Mott Energy, LLC., (LME) shall perform geologic mapping of each foundation excavation for safety-related structures. For a mat resting on drilled deep foundation elements supporting safety-related structures, LME shall perform geologic mapping of exposed portions (where safely accessible) and shall perform geologic logging of soil cuttings obtained from portions constructed under slurry or temporary casing, or otherwise not accessible for direct observation. LME shall examine and evaluate the geologic features and conditions identified for consistency with the site characterization described in Preliminary Safety Analysis Report section 2.5. Once geologic mapping and logging information from the foundation excavation and deep foundation elements for safety-related structures are available for examination by the NRC staff, LME shall notify the Director of the Office of Advanced Reactors, or the Director's designee, as specified in Title 10 of the Code of Federal Regulations (10 CFR) Section 50.4, "Written communications."

APPENDIX B – REFERENCES

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