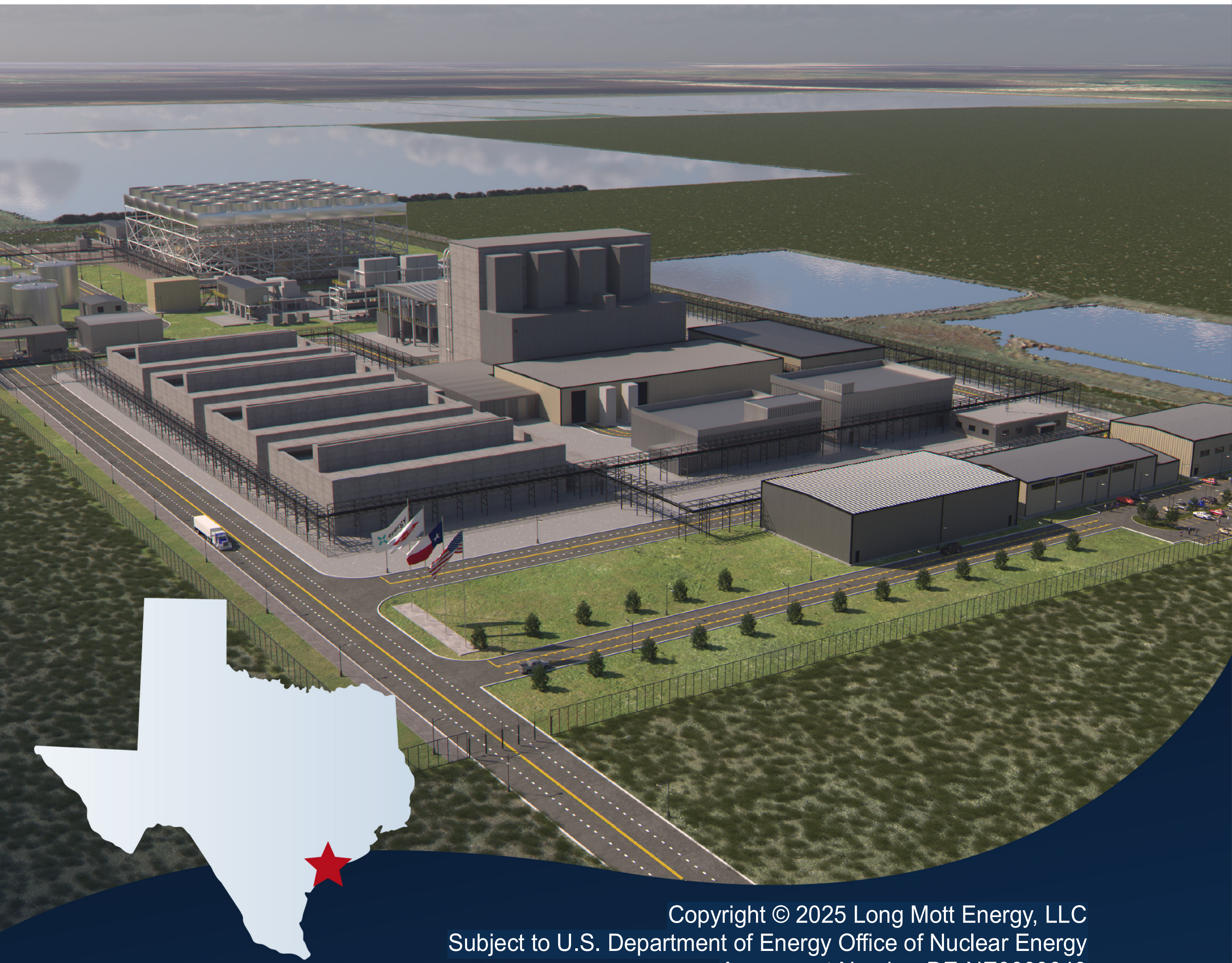
# Long Mott Generating Station

# Construction Permit Application
# Part IV
# Technical Specifications

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## 1.0 Introduction

Pursuant to the requirement of 10 CFR 50.34(a)(5), an identification and justification for the selection of those variables, conditions, or other items which are determined as a result of the preliminary safety analysis and evaluation to be probable subjects of technical specifications (TS) for the facility, with special attention given to those items which may significantly influence the final design are discussed below.

The proposed technical specifications reflect the preliminary design and associated preliminary safety analyses. Table IV-2 and Table IV-3 identify, and provide a justification for, the proposed variables and conditions associated with the development of Safety Limits (SLs) and Limiting Conditions for Operation (LCOs.) The full scope of technical specifications will reflect the LMGS final design and final safety analyses to support the Operating License Application (OLA).

In accordance with the U.S. Nuclear Regulatory Commission (NRC) interim staff guidance (ISG) DANU-ISG-2022-08, “Risk-Informed Technical Specifications” [Reference IV-1], the proposed TS include the following categories:

- Use and Application
- Safety Limits and Limiting Safety System Settings
- Limiting Conditions for Operation
- Surveillance Requirements
- Design Features
- Administrative Controls

The TS will use the format from NUREG-1431, Volume 1, Revision 5, “Standard Technical Specifications” [Reference IV-2]. Detailed Technical Specifications, including applicable bases, will be provided in the Operating Licenses Application (OLA) in accordance with 10 CFR 50.34(b)(6), 10 CFR 50.36(a), and 10 CFR 50.36(b).

### 1.1 Operating Modes

The unit-operating modes, as shown in Table IV-1, encompass the entire envelope of operation. An operating mode corresponds to any one inclusive combination of systems, structures, or components (SSCs) configuration, average primary pressure, and fuel pebble status in the reactor vessel.

#### 1.1.1 Mode 1 - Power Operation

Mode 1 is entered from Mode 2 during startup when all shutdown rods are fully withdrawn and any one of the nine control rods is not fully inserted.

#### 1.1.2 Mode 2 - Startup

Mode 2 is entered from Mode 3 when any of the shutdown rods is not fully inserted and all control rods are fully inserted.

#### 1.1.3 Mode 3 - Shutdown

Mode 3 can be entered from either Mode 4, when primary loop pressure exceeds 0.2 MPa (29 psi), or entered from Modes 1 or 2, when both shutdown and control rod banks are fully inserted with the exception of one rod in each bank not fully inserted and trip switch is in shutdown.

#### 1.1.4 Mode 4 - Depressurized

Mode 4 can be entered either from Mode 5, when at least one fuel pebble is in the reactor vessel and at least 16 of any combination of shutdown or control rods are fully inserted, or entered from Mode 3 when primary loop pressure reduces to less than or equal to 0.2 MPa with any combination of at least 16 rods are fully inserted.

#### 1.1.5 Mode 5 - Defuel

Mode 5 is entered when there are no fuel pebbles in the reactor.

## 2.0 Preliminary Technical Specifications

### 2.1 Use And Application

The Use and Application section of the technical specifications provides definitions, defines the plant modes used in determining Limiting Condition for Operation (LCO) applicability, explains the meaning of logical connectors, establishes the completion time convention and provides guidance for its use, and defines the proper use and application of frequency requirements. The full definitions of this section will be developed to support the OLA.

### 2.2 Safety Limits and Limiting Safety System Settings

#### 2.2.1 Safety Limits

10 CFR 50.36(c)(1)(i)(A) defines SLs for nuclear reactors as the limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity.

The proposed SLs are developed using the methodology in NEI 18-04 and guidance in DANU-ISG-2022-08. The specific limits of the process variables for the proposed SLs will be provided in the TS to support the OLA.

The proposed preliminary SLs are summarized in Table IV-2.

#### 2.2.1.1 SL - Specified Acceptable System Radionuclide Release Design Limits

Specified acceptable system radionuclide release design limits (SARRDLs) shall not be exceeded during any condition of normal operation, including the effects of Anticipated Operational Occurrences (AOOs). The SARRDLs are used as an initial condition to the design basis accident (DBA) analyses to ensure required safety function (RSF) 1, “Retain radionuclides in the fuel particles and pebbles” is met.

#### 2.2.1.2 SL - Reactor Pressure Vessel Pressure

To ensure the RPV pressure design limits will not be exceeded under any operating condition, the RPV pressure SL ensures RSF 1.4.1, Maintain Core Geometry, is met to permit the following RSFs:

- Passive removal of heat from the reactor core to the Reactor Cavity Cooling System (RCCS)
- Sufficient insertion of control rods and shutdown rods and inherent reactivity feedback to provide safe shutdown of the reactor during design basis events (DBEs) and design basis accidents (DBAs)

The RPV pressure limit will be provided in the TS for the OLA.

#### 2.2.1.3 SL - Reactor Pressure Vessel Temperature

To ensure the RPV temperature design limits will not be exceeded under any operating condition, the RPV temperature SL ensures RSF 1.2.1, Maintain Core Heat Removal through Passive Means, is met so that adequate heat is removed from the core to ensure fuel and radionuclide release limits are not exceeded during DBEs and DBAs. The RPV temperature limit will be provided in the TS for the OLA.

### 2.2.2 Limiting Safety System Settings (LSSSs)

10 CFR 50.36(c)(1)(ii)(A) defines LSSSs as the settings for automatic protective devices related to those variables having significant safety functions. Where a LSSS is specified for a variable on which a SL has been placed, the setting must be chosen so that automatic protective action will correct the abnormal situation before an SL is exceeded.

In accordance with the NEI 18-04 methodology and DANU-ISG-2022-08, the proposed preliminary LSSSs are defined as follows:

Settings for automatic protective devices related to those variables that prevent or mitigate a release of radioactive material or protect one or more barriers to maintain the consequences of one or more DBEs or the frequency of one or more high-consequence BDBEs inside the Frequency-Consequence Target. Where a LSSS is specified for a variable on which a SL has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before an SL is exceeded.

To protect SLs, Analytical Limits (ALs) are defined. An AL is the limit of a measured or calculated variable established by the safety analysis to ensure that a SL is not exceeded. Setpoints are predetermined process values for actuation of protective devices. Setpoints ensure that trips or actuations occur at or before ALs in order to ensure SLs are not exceeded. The ALs will be provided in the TS for the OLA.

## 2.3 Limiting Conditions for Operation (LCOs)

The regulations set forth in 10 CFR 50.36(c)(2) require the TS to include Limiting Conditions of Operation (LCOs). LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. They are derived from the safety analysis and risk-informed evaluation.

The NEI 18-04 process specifies that the TS for risk-significant SSCs be consistent with achieving the necessary safety function outcomes for the risk-significant LBEs. Using the NEI 18-04 methodology, SSCs are screened using four criteria provided in DANU-ISG-2022-08 to determine whether an SSC should have an LCO.

TS LCOs for the OLA will use the format in NUREG-1431, Volume 1, Revision 5. For each LCO, the modes of applicability, entry conditions, action statement(s), and completion time will be provided in the TS for the OLA. The proposed preliminary systems subject to TS LCOs considering the preliminary safety analysis report (PSAR) insights are provided in Table IV-3 along with the assessment of the variables/conditions, and the corresponding basis.

## 2.4 Surveillance Requirements

In accordance with 10 CFR 50.36(c)(3), surveillance requirements (SRs) are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within SLs, and that the LCOs will be met.

SRs for LMGS are developed using the methodology in NEI 18-04 and the guidance in DANU-ISG-2022-08.

For each LCO, appropriate SRs will be provided in the OLA. Their frequency of performance will be determined and controlled by a Surveillance Frequency Control Program (SFCP), which will be specified in the Administrative Controls Section of TS.

## 2.5 Design Features

In accordance with 10 CFR 50.36(c)(4), design features to be included in the TS are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered by LCOs.

The design features are determined using the NEI 18-04 methodology and the guidance in DANU-ISG-2022-08. The preliminary proposed design features are provided below.

### 2.5.1 Reactor Systems and Structures

#### 2.5.1.1 Reactor Pressure Vessel

The RPV is designed to support the RSFs of control heat removal and maintain core geometry. The RPV controls heat removal by passively transferring heat from the fuel, through the core metallic structures and graphite core assembly, to the reactor cavity, reactor cavity cooling system, and environment. The reactor pressure vessel is also designed to maintain core geometry by limiting RPV stress to acceptable levels.

#### 2.5.1.2 Graphite Core Assembly

The graphite core assembly is designed to perform RSF 1.1.2.1, Provide Means to Insert Negative Reactivity to Ensure Safety Shutdown; RSF 1.2.1.1, Transfer Heat from Graphite Core Structures; and RSF 1.4.1.3, Maintain Acceptable Graphite Core Assembly Geometry. The graphite core assembly is designed to control reactivity with inherent reactivity feedback and provide sufficient shutdown margin for safe shutdown, such as via RSF 1.1.2.1. The graphite core assembly is designed to control heat removal by passively transferring heat from the fuel to the ultimate heat sink, and to maintain core geometry by limiting core reflector graphite stress to acceptable levels.

#### 2.5.1.3 Core Metallic Structures

The core metallic structures are designed to support RSF 1.2.1.1, Transfer Heat from Core Metallic Structures to Vessel; RSF 1.4.1.2, Limit Stress in Core Metallic Structures; and provide sufficient margin for safe shutdown.

#### 2.5.1.4 Defuel Chute

The Defuel Chute provides the path for fuel pebbles from the reactor core to the fuel handling system (FHS). Fuel enters the FHS after approximately seven days of residence in the defuel chute, allowing the pebbles to cool to below 250°C before entering the FHS.

### 2.5.2 Fuel and Functional Containment (High-Assay Low-Enriched and Low-Enriched Uranium Pebble)

The performance of the fuel is monitored by SARRDL as discussed in 2.2.1.1.The Low-Enriched Uranium Pebble (LUP) and High-Assay Low-Enriched Uranium Pebble (HUP) are designed to perform the retain radionuclides in the fuel particles and pebbles RSF as part of the functional containment to assure the functional containment design limits are not exceeded during DBEs and DBAs.

The HUPs and LUPs are designed to support the control reactivity RSF by providing sufficient inherent reactivity and to perform the control heat removal RSF by transferring sufficient heat via conduction to the graphite pebble matrix to ensure that fuel and radionuclide release limits are not exceeded during a DBE or DBA. The fuel pebble is designed to transfer sufficient heat to the reactor core via conduction, radiation, and convection to assure that fuel and radionuclide release limits are not exceeded during a DBE or DBA.

The HUPs and LUPs are designed to support the maintain core geometry RSF by withstanding stresses developed during DBEs and DBAs to ensure acceptable geometry is maintained to control reactivity and heat removal.

### 2.5.3 Spent Fuel Storage System

The Spent Fuel Storage System (SFSS) provides on-site, shielded storage for irradiated fuel pebbles used over the lifetime of the plant for all units. Spent fuel pebbles are transferred from the FHS discharge gate into the spent fuel canisters in the fuel handling building. The remotely operated special tool (ROST) then picks up the spent fuel canister with the lid attached and transports the canister to the canister processing facility (CPF). The canister will be inspected, decontaminated, and the lid welded on at the CPF. The ROST will then transport the canister from the CPF through the Inter-Unit access tunnel (IUAT) into the Spent Fuel Interim Storage Facility (SFISF) for final on- site storage. The SFISF uses natural circulation air cooling to maintain fuel canister and concrete temperatures below applicable limits.

## 2.6 Administrative Controls

In accordance with 10 CFR 50.36(c)(5), administrative controls are the provisions relating to organization and management, procedures, record keeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

The TS administrative controls are developed using the NEI 18-04 methodology and the guidance in DANU-ISG-2022-08 and address the following areas:

1. The operations organization structure, including roles and responsibilities
2. The description of the onsite and offsite organizations
3. The description of facility staff qualifications
4. The requirement to establish, implement, and maintain procedures
5. The requirement to establish, implement, and maintain programs and reports necessary to operate the plant in a safe manner. Applicable plant programs and reports for consideration include but are not limited to the following:
    - Setpoint control program
    - Surveillance frequency control program (SFCP)
    - Risk-managed technical specifications (RMTS) program, including risk-informed completion time (RICT) and risk management actions times (RMATs)
    - High radiation area controls in accordance with 10 CFR 20.1601(c)
    - Offsite dose calculation manual
    - Radiological effluents control program

- Annual radiological environmental operating report and radioactive effluent release reports
- TS bases control program

The administrative controls, programs, and reports will be provided in the TS for the OLA.

## References

IV-1 U.S. NRC, March 2024, DANU-ISG-2022-08, “Advanced Reactor Content of Application Project, 'Risk-Informed Technical Specifications”

IV-2 U.S. NRC, September 2021, NUREG-1431, “Standard Technical Specifications,” Revision 5, Volume 1.

## Tables

### *Table IV-1: Operating Modes*

| Mode | Title | Fuel Status | Rod Status | Primary Pressure |
|---|---|---|---|---|
| 1 | Power Operation | 1 or more fuel pebbles in the reactor vessel | • All shutdown rods fully withdrawn<br>• Any control rod not fully inserted | Primary loop pressure > 0.2 MPa |
| 2 | Startup | 1 or more fuel pebbles in the reactor vessel | • Any shutdown rod not fully inserted<br>• All control rods fully inserted | Primary loop pressure > 0.2 MPa |
| 3 | Shutdown | 1 or more fuel pebbles in reactor vessel | • At least 16 rods fully inserted | Primary loop pressure > 0.2 MPa |
| 4 | Depressurized | 1 or more fuel pebbles in reactor vessel | • At least 16 rods fully inserted | Primary loop pressure ≤ 0.2 MPa |
| 5 | Defueled | No fuel pebbles in the reactor vessel | N/A | N/A |

### *Table IV-2: Proposed Preliminary Safety Limits*

| Safety Limit | Description | Required Safety Function (RSF) | Basis |
|---|---|---|---|
| 2.2.1.1 Specified Acceptable Radionuclide Release Design Limits (SARRDL) | Plate-out and circulating radionuclide activity inside the Helium Pressure Boundary (HPB) shall not exceed an upper bound for dose equivalent I-131 and Xe-133, respectively, during normal operations and following AOOs. | Retain Radionuclides in Fuel Particles and Pebbles, and control reactivity | SARRDL maintains activity levels in the HPB during normal operations and following AOOs. |
| 2.2.1.2 Reactor Pressure Vessel (RPV) Pressure | RPV internal pressure shall not exceed an upper bound under any operating condition. | Maintain Core Geometry | RPV internal pressure is limited to a value that ensures core geometry is maintained. |
| 2.2.1.3 RPV Temperature | RPV temperature shall not exceed an upper bound under any operating condition. | Control Heat Removal | Adequate heat removal is provided to ensure acceptable fuel and radionuclide release limits are not exceeded during DBEs and DBAs. |

### *Table IV-3: Proposed Preliminary Systems Subject to LCOs (Sheet 1 of 3)*

| No. | Systems | Variable or Condition | Basis |
|---|---|---|---|
| 1 | Reactor Protection System (RPS)<br>Nuclear Instrumentation System (NIS) | 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 protective actions.<br>Initial LCOs trip variables:<br>• Reactor trip - High Reactor Flux, High Startup Rate, Low Helium Pressure Boundary (HPB) Pressure, Low HPB Flow, High Hot Helium<br>• Helium Circulator System (HCS) trip - High Reactor Flux, High Cold Helium, High HPB Pressure, Low Feed Water Flow<br>• Steam Generator System (SGS) isolation - High Moisture, N16 in HPB | The Reactor Protection System (RPS) and Nuclear Instrumentation System (NIS) detect abnormal conditions and perform Required Safety Functions to mitigate licensing basis events (LBEs) or prevent LBEs from occurring. The parameters, including tolerances, which instruments must be capable of detecting to identify abnormal plant conditions to ensure LSSSs are not exceeded will be determined in the setpoint control program. The RPS contains internal RTDs, separate from NIS instrumentations, to measure cabinet temperatures which can be used to detect, and indicate initiating events such as fire, where necessary, a significant abnormal degradation of barriers, which can be used to mitigate DBEs to within the F-C target and DBAs. The RPS supports the following RSFs, which are required to be fulfilled during specific LBEs and support of fuel pebbles RSF 1 [RSF 1.1.2 (Maintain long-term subcriticality) associated with Reactor trip, RSF 1.1.1 (Control reactivity with inherent reactivity feedback) associated with HCS and Reactor Trip, and RSF 1.3.1 (Isolate water/steam sources) associated with SG isolation, HCS and Reactor Trip. In addition, NIS also performs risk-significant NSRST function.] |
| 2 | Investment Protection System (IPS) | The Investment Protection system (IPS) detects abnormal conditions and perform Required Safety Functions to mitigate licensing basis events (LBEs) or prevent LBEs from occurring. The parameters, including tolerances, which instruments must be capable of detecting to identify abnormal plant conditions to ensure LSSSs are not exceeded will be determined in the setpoint control program.<br>Initially identified LCOs variables will be developed for the IPS that define the lowest functional capability of:<br>• Field transmitters and process sensors<br>• Components that provide signal conditioning and setpoint comparison<br>• Components that perform logic functions | LCOs for this system will ensure that the IPS can preform the lowest functional capability for safe operation of the plant and DID for the RPS. IPS performs a risk-significant NSRST prevention function. |
| 3 | Reactor Cavity Cooling System (RCCS) | The Reactor Cavity Cooling System (RCCS) provides passive heat transfer from the RPV to the atmosphere.<br>Initially identified LCOs consists of:<br>• Boil-off head tank level | Sufficient RCCS water inventory is necessary to maintain consequences within the F-C Target and mitigate DBAs. LCOs for this system will ensure that the RCCS can preform the lowest functional capability for safe operation of the plant. The RCCS supports the following RSF, which is required to be fulfilled during specific LBEs, RSF 1.2.1 (Maintain core heat removal through passive means). |
| 4 | Steam Generator System (SGS) | The Steam Generator System (SGS) transfers heat from the primary helium leaving the reactor core to the water/steam flowing through the SG tubes to produce super-heated steam.<br>Initially identified LCOs variables:<br>• Main Steam Temperature<br>• Main Steam Pressure<br>• Feedwater Temperature<br>• Feedwater Pressure<br>• HSS Moisture<br>• Helium to Feedwater Mass Flow Ratio<br>• Isolation valves status | The SGS, via its subsystems support RSF 1.3.1 (Isolate Water/Steam Sources). The isolation of the SGS is accomplished by the closure of Main Steam and Feedwater isolation valves. |

## *Table IV-3: Proposed Preliminary Systems Subject to LCOs (Continued) (Sheet 2 of 3)*

| No. | Systems | Variable or Condition | Basis |
|---|---|---|---|
| 5 | Reactivity Control & Shutdown System (RCSS)<br>Reserve Shutdown System (RSS)<br>Reactivity Control System (RCS) | The Reactivity Control and Shutdown System (RCSS) provides a secondary, independent, and diverse means of shutting down the reactor and maintaining safe shutdown during normal operation, AOOs, DBEs, and DBAs. The RCSS consists of two separate rod banks, the Reserve Shutdown System (RSS) and the Reactivity Control System (RCS), each consisting of nine absorber rods. During normal operations, the RCS is used to control reactor outlet temperature.<br>• Initial identified LCOs variables:<br>• Minimum number of rods per bank | RCSS performs RSF to mitigate the consequences of DBEs to within the F-C target. LCOs for the RCSS system will ensure the RCSS can perform the lowest functional capability for safe operation of the plant. The RCSS supports RSF 1.1.2 (Maintain long-term subcriticality), which is required to be fulfilled during specific LBEs. |
| 6 | Reactor Pressure Vessel (RPV)<br>Defuel Chute Pressure Vessel (DCPV) | The safety-related Reactor Pressure Vessel (RPV), Defuel Chute Pressure Vessel (DCPV).<br>Initial identified LCOs variables:<br>• RPV temperature<br>• RB Temperature<br>• FHS isolation valves status | The RPV and DCPV provide a barrier to maintain release of radioactive materials to within the DBE F-C Target and to mitigate DBAs. They include design features, and operating restrictions concerning limiting stress to acceptable levels, retaining radionuclides, maintaining pressure integrity during transients, and preventing loss of pressure integrity for AOOs and DBEs. LCOs for related systems will ensure they are adequately protected. |
| 7 | Primary Pressure Relief System (PPRS) | The Primary Pressure Relief System (PPRS) provides overpressure protection for the helium pressure boundary (HPB) The PPRS comprises a combination of safety relief valves (SRVs), three-way valves, a power operated relief valve (PORV), and rupture discs connected by piping.<br>Initial identified LCOs variables:<br>• SRVs setpoints based on HPB Pressure<br>• Number of SRVs available | The PPRS valves automatically lifts at HPB pressures to maintain consequences of SGTL DBEs and SGTR DBAs. LCOs for the PPRS system will ensure PPRS can preform the lowest functional capability for safe operation of the plant. The PPRS supports RSF 1, the retention of Radionuclides in Fuel Particles and Pebbles, via RSF 1.4, RSF 1.4.1 (Maintain core geometry), which is required to be fulfilled during specific LBEs. |
| 8 | Reactor Building (RB) | The Reactor Building (RB) is a group of independent above-grade structures comprising four SR steel-frame Reactor Support Structures (RXSTs) that support each reactor and associated components, along with four separate non-safety related with special treatment (NSRST) steel frame Steam Generator Support Structures (SGSTs) supporting the steam generators (SGs) and an NSRST Shield Structure (SST) surrounding, and structurally separate from, the steel frames.<br>Initial identified LCOs conditions:<br>• Nuclear Island Heating, Ventilation, and Air Conditioning (NIHV) available<br>• Reactor Cavity Cooling System (RCCS) available | During normal operations the NIHV and the active portions of the RCCS provide heat removal. All four RXSTs are supported by a common SR reinforced concrete mat supported by concrete piers. The SST and SGSTs are supported by a separate, unconnected NSRST foundation. The RXSTs support performance of RSF 1.4.2 (Maintain reactor support geometry), which is required to be fulfilled during specific LBEs: |

**Table IV-3: Proposed Preliminary Systems Subject to LCOs (Continued) (Sheet 3 of 3)**

| No. | Systems | Variable or Condition | Basis |
|---|---|---|---|
| 9 | Helium Circulator System (HCS) | The Helium Circulator System (HCS) circulates helium through the primary system as a means of transporting thermal energy from the reactor core to the steam generator, where it is then transferred to the secondary system for power production, process heat utilization, or residual heat removal. The HCS functions as part of the HPB to maintain primary system pressure and facilitate heat removal from the reactor core.<br>Initial identified LCOs variables:<br>• Helium to Feedwater Mass Flow Ratio<br>• Number of circulators available | HCS performs risk-significant NSRST function and the LCOs for the HCS will ensure HCS can preform the lowest functional capability for safe operation of the plant. The HCS supports PSF 1.2.2 (Maintain core heat removal through active means), and PSF 2.7 (Maintain HPB pressure integrity). |
| 10 | Fuel Handling System (FHS) | The Fuel Handling System (FHS) includes the infrastructure and interfaces with the HPB as required to load and unload fuel pebbles into and out of the reactor. The FHS conveys the fuel pebbles by means of gravitational and pneumatic forces using helium as a transport gas supplied from the HSS.<br>Initial identified LCOs variables:<br>• Fuel burnup measurement<br>• Pebble dimension check | FHS performs risk-significant NSRST function and the LCOs for the FHS will ensure FHS can preform the lowest functional capability for safe operation of the plant. The FHS supports PSF 2 (Retain radionuclides in the HPB). |