**U.S. Nuclear Regulatory Commission Staff Observations on Terrestrial Energy USA, Inc.'s Integral Molten Salt Reactor Core-Unit Definition White Paper, Revision 1**

By letter dated May 6, 2022, (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML22138A340), Terrestrial Energy USA, Inc. (TEUSA) submitted a white paper titled, "IMSR®[Integral Molten Salt Reactor] Core-unit Definition – Applicable Structures, Systems and Components – Revision 1." TEUSA requested the U.S. Nuclear Regulatory Commission (NRC) staff to review the white paper and provide written feedback and questions on areas where the white paper does not provide sufficient detail for the NRC staff to have a clear understanding of the design basis and operation of the IMSR power plant, and more specifically on those structures, systems, and components (SSCs) that are important to the safe operation of the IMSR Core-unit. The NRC staff concludes that the white paper provided sufficient information for the NRC staff to understand the general design basis and operation of the IMSR. However, while assessing the white paper, the NRC staff did identify areas that should be considered during the development of a core-unit standard design approval (SDA) application, or information that would likely be useful during a regulatory review of related systems in future licensing applications such as part of a design certification application or topical report. It should be noted that not all of the information would necessarily be required on the docket of an application but could instead be addressed during an audit of supporting information correlating to high level information that is on the docket. In the feedback provided below, any information which the NRC staff felt would likely be necessary to directly include in an SDA application is stated as such. Information which could addressed in other licensing activities (e.g. Part 50 or 52 application, topical report application, etc.) or potentially as information likely requested in an audit supporting such licensing activities is simply referred to as "in future licensing applications".

**Note: [[ ]] denotes proprietary information.**

**Feedback that should be considered for future submittals, including Standard Design Approval (SDA) application:**

1) **Section IV Silos and Reactor Vaults**

   a) The silos and reactor vaults appear to interact with the core-unit through a physical means via the reactor support structure. Does TEUSA intend to include a seismic analysis of the core-unit in the SDA application? If so, then the seismic response of the silos and Rx vaults needs to be considered and be available during review of the SDA application.

2) **Section IV Salt Leakage Detection**

   a) The described method of **[[**[redacted]**]]** lacks information on sensitivity. Future licensing applications should expand on how fuel salt level detection would function **[[**[redacted]**]]** and how the accuracy of that leakage detection method would provide prompt detection of fuel salt in the **[[**[redacted]**]]**.

   b) The described secondary method of leakage detection states that it is provided for all three loops, but the description is not adequate to provide an understanding of how the leakage detection would function for the primary fuel salt loop. Specifically, include in future licensing applications, a description of how the [[redacted]].

**3) Section IV Reactor Support Structure**

a) Does TEUSA intend to include a seismic analysis of the core-unit in the SDA application? If so, the response spectrum of the reactor support structure during a seismic event (or any motion-based event including those of a dropped crane, if applicable) should be available during the SDA application or otherwise considered in future licensing applications.

**4) Section IV Containment**

a) This section describes [[redacted]]. Consistent with the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.44(d), describe in future licensing applications whether combustible gas generation is technically relevant to the IMSR design.

b) This section describes [[redacted]]. Describe provisions for containment integrated leakage testing in future licensing applications.

**5) Section IV Pipe Journals**

a) In future licensing applications, describe how the [[redacted]].

**6) Section IV Hot Cell**

a) Other sections in the paper describe [[redacted]] but it is not mentioned in this section. Is this an oversight?

**7) Section IV Purge and Pressure System**

a) Clarify the method of containment isolation during normal operation and the capability to test isolation devices for leakage.

**8) Section IV Internal Reactor Vessel Auxiliary Cooling System**

a) In the internal reactor vessel auxiliary cooling system (IRVACS), the decay heat is removed from the internals of the core-unit and guard vessel walls by convection and/or radiation. Is there an additional boundary layer of separation from the reactor containment and the air-cooled emergency heat exchanger (Figure 9)?

b) A detailed discussion addressing the interactions between IRVACS and the core-unit will be needed to support any future core-unit SDA applications. Some areas that the NRC staff might cover in greater detail in its review of a core-unit SDA application include:

i) IRVACS 1a system to confirm that no failure of the pump or fan (e.g. a potential locked rotor event) could prevent the passive performance of the loop.

ii) The [[redacted]] which is used to control thermal performance of the IRVACs to determine if there is a failure mode which could result in unstable performance (e.g. corrosion of the device, etc). If this is the case, the core-unit SDA application review will cover core performance in an unstable thermal performance situation.

iii) The method by which the IRVACS is connected to the core-unit and whether or not there is a single failure which could result in failure of the entire IRVACS system (e.g. type of welding used, whether or not an common shared equipment exists (e.g. manifold), etc).

c) Please be consistent with either metric or imperial scale in the report. For example, [[redacted]].

**9) Section IV Irradiated Fuel Auxiliary Cooling System**

a) The irradiated fuel auxiliary cooling system (IFACS) appears to be closely related to IRVACS, at least conceptually. There are fewer figures and less information in general, describing the IFACS. The NRC staff would likely have similar questions about the IFACS as the IRVACS in future licensing applications but, it is recognized that the role of IFACS in the safety case might be different.

b) How many thermal static valves (TSV) are available? Is TSV considered safety-related? Is there a redundancy of passive control flow available in case of TSV malfunction?

**10) Section IV Secondary Coolant System**

a) What is the pressure differential between the primary and secondary side? How much time before the pressures equalize in the event of a leak in a primary heat exchanger tube?

**11) Section IV Tertiary Coolant System**

a) [[redacted]]

**12) Section IV Irradiated Fuel System**

a) The section states "At the [[redacted]], the IrFS has provisions for [[redacted]]." However, the NRC staff has not identified a general description of how these actions would be accomplished. A general description of how this action would interface with the core-unit would be needed to support the core-unit review.

13) Section IV Irradiated Fuel Salt and Gas Transfer System

a) The section includes the following description: "As the FSST [fuel salt storage tanks] and solidified fuel salt cool, the FSST will thermally contract more than the fuel salt, as **[[**[redacted]**]]** has a higher thermal expansion rate than the fuel salt. Depending on the structural properties of the fuel salt and the FSST, this may cause deformation of the FSST as it contracts around the fuel salt." Please explain the code or standard that would be used for design of the FSST (and any associated guard vessel) and how plastic deformation conforms with the standard.

b) Please provide the basis for concluding radiological and neutronic conditions that would support continued operation of the fuel salt without a **[[**[redacted]**]]**.

14) Section IV Gas Holding Tank Cooling System

a) The reliability target and code classification specified for the gas holding tank cooling system will need to be justified with additional information, including the radiological consequences of a tank leak and possibly the likelihood of tank overpressure/leakage, if cooling is not available.

15) Section IV Hot Cell Cooling System

a) What will the coolant (or options for coolants) be?

16) Section IV FSST Vault Cooling System

a) The FSST Vault Cooling System section lists three reasons for providing cooling requirements. The first two stated reasons do not provide associated quantitative values. If these limits are necessary in order to analyze the core-unit performance, then include these values within the future core-unit SDA application.

17) Section IV Process Reactor Auxiliary Cooling System

a) While it is recognized that the design is currently evolving, the NRC staff will expect to see discussions in a future core-unit SDA application regarding: (1) physical attachment of the Process Reactor Auxiliary Cooling System and compliance with any applicable codes, (2) logic used to begin use and maintain temperature, and (3) consequences of any failed operation including overcooling.

18) Section V Instrumentation and Control

a) Figure 30: Please include units on all numbers.

b) Figure 31: What are the roles of **[[**[redacted]

[redacted]]].

c) Figure 32. Why are the lines dashed and not solid? Again, why do some lines not complete a path? If lines cross but do not interact, please indicate so.

d) The maximum temperature demands on the instrumentation listed appears to be outside of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III, Division 5, ranges. Provide supporting justification for any such equipment requirements.

**19) Section V Pump Vibrating Monitoring**

a) Is TEUSA following any NRC-endorsed guidance pertaining to pump vibration monitoring systems? For example, NUREG-0800, "Standard Review Plan" (SRP), Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components," provides guidance to the NRC staff in reviewing dynamic testing and analysis of piping systems, mechanical equipment, reactor internals, and their supports under vibratory loadings. While it is written for light-water reactors (LWRs), the information can still be useful for non-LWRs.

**20) Section V Leak Detection**

a) The information provided in the white paper indicates that TEUSA is aware of the leak detection needs that are proposed in draft design criteria TEUSA-14 per the principal design criteria white paper. Additional information will be needed on the capabilities of the leakage detection equipment, associated instrumentation and controls, and the material compatibility of the detectors and the environment.

b) The white paper states, [[redacted]]. Are the methods for detecting the conditions different during operation versus during the core swap

**21) Section V Environmental Qualification Envelope**

a) The information provided is high level and qualitative due to the design level at this time. The NRC staff has no comments at this time. The NRC staff's review as part of future licensing applications will cover, in detail, the qualification methods for the necessary environments, especially given the high temperature and radiation levels in some locations.

**22) Section VII Out-of-core Criticality**

a) The possibility of out-of-core criticality is unclear. It is both stated that "out-of-core criticality cannot occur" and "criticality alarms will be installed in areas where it is possible for criticality to occur". Clarify this apparent discrepancy and identify which areas will have alarms.

**23) Section VII Reactor Thermal Hydraulics**

a) On page 93 of the white paper, it is stated "[t]he thermal hydraulics simulations and analysis are performed based on the proven practices for validation and verification, using acceptable codes and standards, and operating experience from other operating reactors". Which codes and standards and operating reactor experience does this refer to?

b) On page 93, it is stated that thermal hydraulic analysis and simulations are performed independently using two different codes. The results from these codes are used to perform code-to-code validation. It is not clear to the NRC staff if either of these codes have been (or will be) validated against experimental data. If neither have been validated against experimental data, then comparing the results of the codes would not serve as a validation. Include additional details regarding verification and validation of all codes and methods used in the safety analyses as part of a core-unit SDA application.

c) In Table 4, what is the uncertainty range of the simulated values? Were pressure and void fraction calculations under steady-state operating conditions also used to support the design of SSCs? Are there similar transient calculation simulations that are used to calculate the limiting conditions to which the SSCs must be designed to handle?

**24) Section VII Liquid Fuel Salt**

a) Additional information will be required regarding the methods for collecting fuel salt physical properties and material interaction properties. This information can be made available via audit to support the high-level information presented in the application.

**25) Section VII Pumps**

a) Future licensing applications should address any pump inspection requirements and consequences of any pump failures.

b) The instrumentation and control section indicates that pump instrument types are still under consideration. Any pressure taps or other installed equipment to monitor pump performance should be described in the core-unit SDA application.

**26) Section VII Graphite Moderator**

a) Figure 48. Where does **[[**[redacted]**]]**

b) Figure 49. Add more labels. **[[**[redacted]**]]**

**27) Section VII Shielding**

a) What form is the neutron absorber in the thermal shield (e.g. powder that is stored in capsules, metal matrix composite, etc.)?

b) What codes/methods/criteria are used to calculate nuclear lifetime for the thermal neutron shield, flux, fluence, etc.?

c) Is helium buildup a concern in the shielding (i.e. blistering)?

**28) Section VII Upper Hold-down Plate**

a) How is the graphite moderator secured? Does the upper hold-down plate simply **[[**[redacted]**]]** or does it consist of components such as an adapter plate, enclosure, hold-down springs, clamps, and pads?

**29) Section VII Shutdown Rods**

a) Are the shutdown rods seismically qualified? Are they considered safety-related?

**30) Section VII Load Combinations**

a) The white paper describes "national building standards" as being used in seismic response evaluations of the core-unit design. Additional information will be needed to understand what the "national building standards" are and how they relate to NRC requirements. The NRC staff notes that RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analyses," and NUREG-0484, "Methodology for Combining Dynamic Responses," provide guidance regarding load combinations.

**31) Section VII Acceptance Criteria**

a) What service level is mandated for the components related to the shutdown rods? Describe how the chosen service level does not allow for gross plastic deformation which could prevent insertion of the shutdown rods.

**32) Section VII Core-unit Externals**

a) Is the guard vessel affected by the gamma-ray/radiation swelling over the lifetime of the plant? Is the guard vessel monitored or inspected?

**33) Section VIII Start-up**

a) How will the fuel be initially melted? How will the Initial Fuel System interface with the core-unit?

**34) Section VIII Low-Power and Critical**

a) Since the start-up section concludes with reaching criticality, the NRC staff assumes that this section is supposed to be titled simply "Low-Power". Is this correct?

**35) Section VIII High-Power/Full-Power**

a) What is the sequence for shutting down? In a future core-unit SDA application, describe the shutdown states in detail (hot, warm, cold) and what sequence operators would need to follow to safely shut down the reactor.

**36) Section VIII Core-swap**

a) Additional details about the core-swap procedure will be necessary during the core-unit SDA to help the NRC staff understand possible failures that could occur (e.g. could a spent core drop from a crane during maneuvers and land on or otherwise impact the operational core).

b) Additional description about how the robotic manipulators disconnect (and later connect) the core-unit from the secondary side will be necessary during the core-unit SDA application. In particular, the NRC staff will be interested in any potential initiating events that could occur while connecting a core-unit to the secondary side.