TERRESTRIAL
ENERGY USA

Principal Design Criteria

for IMSR® Structures, Systems and Components

Abstract
TEUSA's long term licensing objective is to obtain a Standard Design Approval (SDA) for the IMSR® Core-unit. A necessary component of a 10 CFR Part 52 SDA application for the IMSR® Core-unit is the identification and description of the principal design criteria for the IMSR® structures, systems and components. This white paper contains the principal design criteria for those systems that provide important functions in support of the operation and safety of the IMSR® plant. The principal design criteria are developed based on the key design features of IMSR® technology and the guidance of Revision 0 of Regulatory Guide 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors."

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I Purpose

The purpose of this white paper is to provide a set of principal design criteria (PDC) for the IMSR® systems, structures and components (SSCs) that provide important functions in support of the operation and safety of the IMSR® facility. Upon approval by the NRC, TEUSA intends for these PDC to be referenced in a subsequent application for a Standard Design Approval (SDA) for the IMSR® Core-unit or in future applications for construction of an IMSR® facility.

The General Design Criteria (GDC) in 10 CFR 50, Appendix A provide the minimum requirements for the principal design criteria (PDC) for water-cooled NPPs. The PDC establish the necessary design, fabrication, construction, testing, and performance requirements for SSCs important to safety. SSCs important to safety are those that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

The requirements for the contents of an application for a construction permit specified in 10 CFR 50.34 (a) (3) include submittal of the PDC for the facility. It goes further to describe the information as the design basis and information relative to materials of construction, general arrangements, and approximate dimensions sufficient to provide reasonable assurance that the design will provide an adequate margin for safety. TEUSA licensing strategy is to initially pursue an application for an SDA. As part of the effort to develop the necessary information to incorporate into the future SDA application, TEUSA has determined that the requirements for submittal of PDC in an application for an SDA are identical to the requirements presented in 10 CFR 50.34 (a)(3).

The information requirements for a Core-unit SDA application is a subset of the information requirements supporting an application for a construction permit or combined license, thereby supporting the longer-term licensing goals associated with IMSR® deployment. Information that supports an SDA application for the IMSR® Core-unit includes information identifying, defining, or describing:

- the IMSR® Core-unit,
- the associated Core-unit engineering boundary conditions,
- the interfaces between the Core-unit and the remaining portions of the IMSR® power plant,
- the IMSR® Principal Design Criteria (PDC),
- the Core-unit interface requirements & acceptance criteria, and
- other regulatory requirements applicable to the IMSR® Core-unit.

II Introduction

Terrestrial Energy USA, Inc. (TEUSA) is developing the Integral Molten Salt Reactor (IMSR®) design to provide electricity or process heat to U.S. industrial heat users. TEUSA is planning for the first commercial deployment of this technology in the late 2020s. The IMSR® is a Generation IV advanced reactor power plant that employs a fluoride molten salt reactor (MSR) design. The IMSR® nuclear power plant (I-NPP), consists of a nuclear island containing at least one, approximately 440 MWth IMSR® (IMSR400) Core-unit. The IMSR400 has the potential to generate up to 195 MWe of electrical power or to export 600 °C of heat for industrial applications, or some combination of both. The I-NPP includes an adjacent balance-of-plant building that contains non-nuclear-grade, industry-standard power conversion and generation equipment.

The IMSR® design builds upon pioneering work carried out at Oak Ridge National Laboratory (ORNL) from the 1950s to the 1980s, where MSR technology was developed, built, and demonstrated with two experimental MSRs. The first MSR was the Aircraft Reactor Experiment (ARE) and next, the Molten Salt Reactor Experiment (MSRE). Based on the demonstrated feasibility of MSR technology, ORNL commenced a commercial power plant program for MSR technology. This program led to the Denatured Molten Salt Reactor (DMSR) design in the early 1980s.

TEUSA has developed and submitted a Regulatory Engagement Plan (REP) (Reference 5) to the Nuclear Regulatory Commission (NRC). The REP outlines topics and schedules for interaction with the NRC to achieve early resolution of general technical or regulatory matters related to the IMSR® design. More specifically, the REP highlights technical and regulatory topics that directly support the development and submittal of a 10 CFR 52, Subpart E application for a Standard Design Approval (SDA) of the IMSR® Core-unit. This white paper [ ] support the TEUSA SDA application development efforts.

Company Background

TEUSA [ ]. TEUSA is a Delaware C-Corp founded in August 2014 that started active business operations in 2015. TEUSA is a U. S. majority-owned company with corporate offices in New York. [ ].

Canadian Nexus

TEUSA has [ ]. TEUSA leverages the ongoing engineering and regulatory work that TEI accomplishes as TEI currently advances its regulatory activities under Phase 2 of the Vendor Design Review (VDR) process with the Canadian Nuclear Safety Commission (CNSC). Leveraging the efforts of TEI's VDR activities is possible because most of the technical and engineering information used for both regulatory reviews is the same. Leveraging TEI effort eliminates duplicate technical work in the U.S., and the approach also provides substantial cost savings for TEUSA. The figure below provides [ ].

Figure 1: [ ]

[

]

Licensing Strategy and Objective

The REP provided to the NRC outlines the regulatory strategy for TEUSA activities. [

] to support a commercial operation date for the first U.S. plant in the 2020s. During regulatory reviews, the NRC uses its understanding of the design and operating characteristics as well as the supporting research and engineering work to perform its review responsibilities efficiently. To support the NRC understanding, TEUSA has begun familiarizing the NRC with the IMSR® design as well as the scope of the available and planned analyses, testing, and operational experience in support of the design. By initiating the process of introducing the IMSR® design information to the NRC, TEUSA expects that the NRC can identify any issues that may require further testing or technical analyses. Additionally, the NRC will be more able to estimate the resource and schedule requirements necessary to conduct regulatory activities associated with IMSR® licensing.

TEUSA's long-term licensing objective for the commercial deployment of the IMSR® design in the U.S. is to first obtain an SDA for the IMSR® Core-unit under 10 CFR Part 52, Subpart E. The IMSR® Core-unit represents a significant technical portion of the IMSR® facility and includes many systems that perform important safety functions. The systems within the Core-unit are reasonably discernible from systems outside the boundaries of the Core-unit. Subsequent sections of this white paper provide additional details about the design envelope of the IMSR® Core-unit and its safety interfaces.

[ ]. The first technical document that TEUSA submitted to the NRC was a white paper that provided an overview of major plant buildings, structures, systems and components that make up the IMSR® facility. This document also provided details of the important plant structures, systems and components (SSCs) that comprise the IMSR® Core-unit. This white paper [ ] identifies a set of principal design criteria (PDC) for the IMSR® structures, systems and components (SSCs) that provide the necessary design, fabrication, construction, testing, and performance requirements for SSCs important to safety.

The TEUSA Regulatory Engagement Plan (Reference 2) provides additional details regarding TEUSA licensing activities and objectives.

III IMSR® Power Plant Description - Overview

Historically, there have been primarily two different types of molten salt reactors that have been developed, were considered for development, or are under development. In one type, solid-fueled reactors use molten salt as a coolant. In the second type, the molten salt also contains the nuclear fuel dissolved in the salt, i.e., the nuclear fuel is also a salt, and the molten salt mixture circulates through a region where nuclear fission occurs to produce heat. In this situation, a reactor is considered a "liquid-fuel" MSR, and this liquid-fuel approach is the basis for the IMSR®.

The power plant described in this paper includes the reactor and power conversion process for creating heat in the reactor core and subsequently transferring the heat to produce electricity. Also included is an overview of the site layout and a brief description of the Reactor Auxiliary Building, Turbine Building, Control Building, the Maintenance Building, and other support buildings.

Reactor and Power Conversion Process

The IMSR® is a liquid-fueled, thermal spectrum, burner-type, fluoride molten salt reactor design that uses standard assay low-enriched uranium fuel, with less than 5% enriched ^{235}U. IMSR® design choices permit the use of liquid-fuel MSR technology in an industrial or commercial setting through simplicity and safety of operation. All of the primary reactor components, including the pumps and heat exchangers, are inside a sealed and replaceable Core-unit with the reactor vessel and its closure head forming the primary boundary of the Core-unit. The result is a simplified reactor plant with no external primary system piping loops, no external primary system pumps, and no pressurizer of any kind. The nuclear fuel and coolant circulate entirely within, never exiting, the reactor vessel. The Core-unit operating lifetime is 7-years. After this period, a new Core-unit replaces the spent Core-unit. This approach eliminates any need to open the Core-unit for graphite replacement, maintenance, or repairs, a complex and costly task made hazardous by potential exposure to radioactivity. The design also provides a high degree of safety and unprecedented simplicity of industrial operation, and by extension, materially lower capital and operating costs compared to other power reactor designs in operation today.

The IMSR® fuel salt is a highly stable, fluoride-based, inert liquid with robust coolant properties and intrinsically high radionuclide retention capabilities that operates at a temperature of approximately 700°C. During normal, critical reactor operations, the primary pumps circulate the fuel salt through the reactor moderator and primary heat exchangers. The liquid fuel salt [

]. The fissioning of the fuel raises the temperature of the fuel salt as it passes through the moderator region. The fuel salt, [
] where it is directed [
]. Near the top of the reactor vessel, [

].

Heat is transferred from the Core-unit via a secondary coolant system, a system using non-radioactive molten salt as the coolant. The secondary coolant system [

] secondary heat exchanger. There are [
]. In the [ ], the secondary coolant

[

]. As the secondary coolant salt passes through the secondary heat exchanger, the heat transfers to a tertiary molten salt coolant loop. After passing through the secondary heat exchanger, the now cooler secondary coolant salt [

].

The Tertiary Salt Loop utilizes an inexpensive, common molten nitrate solar salt. This loop transfers the heat from the secondary heat exchangers in the Reactor Auxiliary Building, to the balance-of-plant building for electricity production, industrial process-heat uses, or both. The tertiary coolant [
]. The Tertiary Salt Loop provides the following:

- Adds an additional barrier and inherent heat sink between the non-radioactive Secondary Coolant Loop salt and the non-nuclear steam plant.
- [

  ].
- [

  ].
- In the event of any tube leakage from the Steam Generator, [
  ], unlike fluoride salts, [
  ].

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1 [

].

The Tertiary Salt Loop is insulated and contains isolation valves on the inlet and outlet pipes for operational and maintenance purposes. The Tertiary Salt Loop design accommodates inspection and maintenance as required, the same as the Secondary Coolant Salt Loop.

Within the balance of plant, the heat output can be used for power generation or process heat, or any combination of power and process heat. For process heat applications, either a steam cycle or direct use of the solar salt is possible. For power generation applications, the Tertiary Salt Loop heats pressurized feedwater in the steam generator, boiling the feedwater to steam under pressure. [

] a commercial, standard high-pressure steam turbine. After passing through the HP turbine, the partially expanded steam [

], the steam enters the intermediate pressure turbine. After exiting the intermediate pressure turbine, the steam flows to low-pressure turbines. After expanding through the low-pressure turbines, the exhaust steam condenses in a condenser. The resulting condensate is sent through feedwater heaters and a deaerator to provide deaerated, [ ] feedwater at the inlet to the steam generator.

The steam turbine drives an electrical generator, generating up to ~195 MWe of turbine island output. The exact net output is site and heat-sink dependent.

Site Overview

The IMSR® site layout includes the buildings required within the site boundary to operate the plant safely and to meet the licensing, safeguards, and security requirements. A typical site has a small footprint (about 7 hectares or 17 acres) and a small security perimeter (approximately 130m x 145m).

An I-NPP site includes a Reactor Auxiliary Building, Turbine Building, Control Building, and Maintenance Building. Also included are the plant support buildings and structures. These include the Administration Building, Simulator and Training Building, Radioactive Waste Building, Coolant Salt Storage Building, Emergency Mitigating Equipment Building, Main Pump House, Water Treatment Building, Fire Water Pump Building, Cooling Water Outlet Building, Sewage Treatment Plant, Electrical Switchyard, and a Security Building.

Figure 2: Typical Site Layout for a Single I-NPP

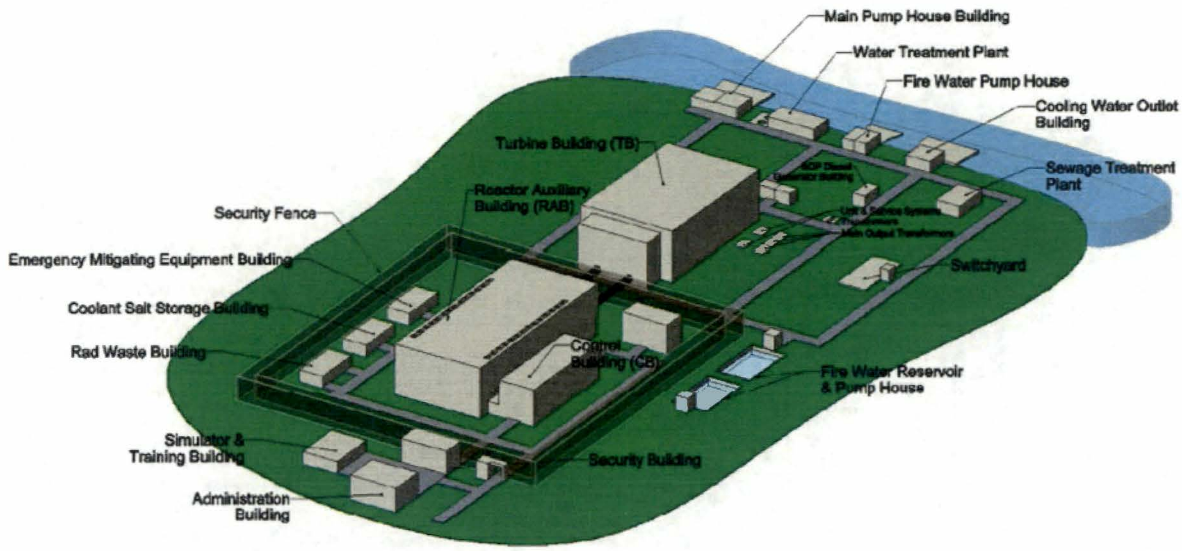


Figure 2 above represents a simplified plant layout identifying the arrangement and configuration of the major buildings, structures, and boundaries of an I-NPP site. A "generic design site envelope," is used to develop the IMSR® site design and encompasses generic site parameters used in Canada, the U.S. and European countries relevant to nuclear plant siting.

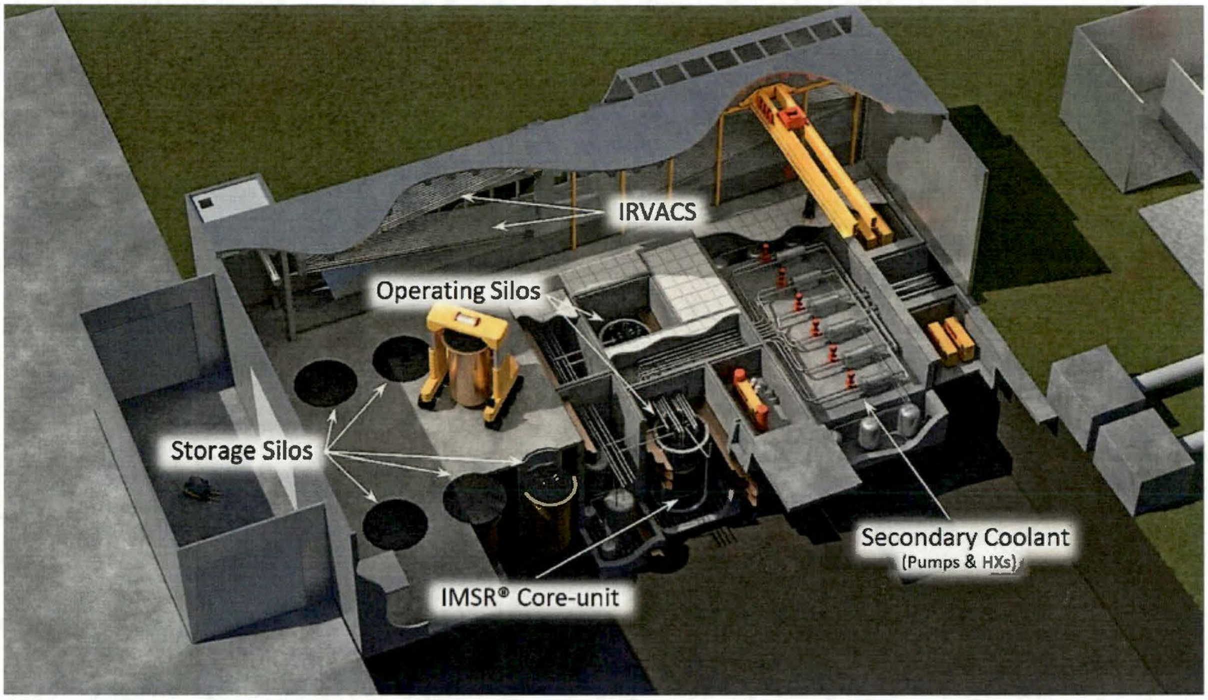
The following section provides, in general terms, a description of the main buildings and structures of a site containing an I-NPP.

Reactor Auxiliary Building

The seismically qualified Reactor Auxiliary Building (RAB) houses, (i) the IMSR® Core-units and associated nuclear systems, (ii) the various heat removal systems before the Steam Generators and, where required, additional heat transfer equipment to supply process heat to industrial users, and (iii) electrical systems and various auxiliary systems required to operate safely, control, and monitor the plant during all postulated operating conditions. Figure 3 below shows the arrangement of the storage silos, operating silos, a portion of the Internal Reactor Vessel Auxiliary Cooling System (IRVACS), and the location of the secondary coolant system; primarily the pumps and heat exchangers. Importantly, Figure 3 shows the physical relationship between the operating silos, storage silos, and the IMSR® Core-unit.

The Reactor Auxiliary Building [ ].

Figure 3: Reactor Auxiliary Building



Turbine Building

The Turbine Building houses the steam generators, steam and feedwater systems, and cooling systems associated with the turbine generator and the supply of electricity to the grid. The specific design requirements for the Turbine Building layout are dependent on the specific turbine generator and the cooling medium selected for condenser cooling.

The steam plant and the associated buildings have no safety function for the I-NPP and therefore are located outside of the protected area. IMSR® employs a conventional industrial electrical generator system with superheated and reheated steam capabilities as well as multi-stage feedwater heating and a condenser unit.

For power generation, the I-NPP uses standardized superheated steam plant equipment such as the steam generators. The plant's steam generators are based on operating experience from various concentrated thermal solar power plants that use similar liquid nitrate salt-heated steam generators to produce steam for powering turbo-generators. The turbine is a power conversion system designed to change the thermal energy of the steam flowing through the turbine into rotational mechanical work, which rotates a generator to provide electrical power.

The Turbine Building [ ].

Control Building

The Control Building houses the main control center, the security and operations staff, associated change rooms, and facilities required for the operation of the plant. [ ] provide for personnel ingress and egress and, for routing of auxiliary, electrical, instrumentation, and communication conduits between the buildings.

The Control Building [ ]. The Control Building [ ].

The Control Building [ ]

Standby Power Buildings

The Standby Power Buildings [ ] safety requirements (i.e., monitoring) [ ] selected equipment. Additional [ ] capability.

The Standby Power Buildings [ ].

Maintenance Building

The Maintenance Building houses the active and non-active shops and facilities associated with the maintenance of equipment removed from the Nuclear Area, as well as the equipment within other buildings and structures.

The Maintenance Building [ ].

IV System and Component Descriptions

Silos

There are eight silos included in the IMSR® facility. Two silos are for operating Core-units, and six are for Core-unit storage. One of either of these two operational silos houses the operating Core-unit for its 7-year operational life; the second silo houses the previously operated (spent) Core-unit during its radioactivity decay cooldown period. Following cooldown, preparations are made for a new Core-unit by transferring the previously operated Core-unit from its operating silo into a storage silo. The six storage silos only house spent Core-units that have completed the required radioactivity decay cooldown period. The silos interface with the Reactor Vessel and Reactor Support Structure. Figure 3 shows the relationship of the Silo to the Guard Vessel, Core-unit, and other systems, structures, and components.

The Silo [ ].

Guard Vessel

The Guard Vessel is a stainless-steel vessel that is fitted around and supports the Reactor vessel. The primary purpose of the Guard Vessel is to catch and retain any fuel salt leakage or radioactive release from the IMSR® Core-unit to protect from any unintended release from the Core-unit. In the event of a Beyond Design Basis failure of the reactor vessel, the Guard Vessel will catch and contain any leaked fuel salt. Unlike the Reactor Vessel, which is a component part of the replaceable Core-unit, the Guard Vessel is a component of the containment boundary. The Guard Vessel is designed to last for the operating life of the plant. Figure 3 shows the relationship of the Guard Vessel, Silo, Core-unit, and other structures, systems and components.

[ ]

The Guard Vessel [ ].

Reactor Support Structure

The Reactor Support Structure is a steel structure located in the silo. The Reactor Support Structure is used to support and provide alignment of the Guard Vessel inside the Silo. By extension, the Reactor Support Structure also provides support and alignment of the Core-unit. Figure 3 shows the relationship of the Reactor Support Structure, Silo, Guard Vessel, Core-unit, and other SSCs.

The Reactor Support Structure [ ].

Containment

The Containment is a metallic structure that that forms a sealed, low-leakage envelope to house all systems that may contain highly radioactive material. Specifically, the containment houses the Reactor Vessel, Fuel Salt Storage Tank(s), and Fuel and Off-gas transfer lines with the Reactor Vault. The Containment operates a a slightly negative pressure so that operating leaks from the Reactor Vault or Reactor Vessel leak into the Containment. In the event of a leak in any of these systems, the containment prevents the release of any radioactive materials to the Reactor Auxiliary Building. The Containment system also minimizes releases in the unlikely event of a severe accident. The Containment system includes the Guard Vessel and a common containment boundary that encloses the top plate of the Reactor Vessel, the off-gas and fuel transfer lines, and the irradiated fuel storage tanks.

The main functions of Containment are to:

- Provide a passive barrier for high activity sources within the plant to protect workers and the public from radiation doses during normal operations and accidents. The main sources of radioactive materials in the plant are the reactor core, off-gas storage, and irradiated fuel systems.
- Control personnel access into Containment to protect plant personnel from radiation. [ ].
- Shield plant personnel working above the Reactor Vessel in the RAB from ionizing radiation.
- Minimize leakage to assure that normal operation release limits are met, and AOOs and DBAs do not result in exceeding dose acceptance criteria defined in TEI Design Guides.

The Containment is continuously sealed when the reactor contains fuel. The Containment environment is conditioned prior to the initial start-up of a Core-unit and then sealed. It does not require [for the time period between Core-unit replacements.

The Containment [ ].

Internal Reactor Vessel Auxiliary Cooling System (IRVACS)

Under normal conditions for reactor heat removal, the fuel salt mixes convectively in the Core-unit and transfers heat out through the primary heat exchangers to the secondary heat exchangers as described above in the Reactor and Power Conversion Process section. The IRVACS functions as an alternate emergency heat sink to passively remove heat generated within the Core-unit during transients, accidents, or whenever the normal heat removal paths are lost.

IRVACS [ ] system that operates [ ] to transfer heat from the Core-unit to the atmosphere. IRVACS is always operating and does not require any AC or DC electrical power. The system functions continuously irrespective of Core-unit status or plant state. The system has no actuation devices, no flow control mechanisms, nor any other type of control device. The heat removal capacity is sized to remove the maximum postulated decay-heat load, including after an accident where normal heat removal might not be available.

IRVACS is seismically qualified, highly robust, and fail-safe. It provides a passive, [ ] to transfer heat from the hot, uninsulated silo to the atmosphere. During all plant operations, the hot Reactor Vessel [ ] Reactor Auxiliary Building. [ ] to the atmosphere. [ ] the process is repeated. Figure 3 shows the relationship of IRVACS, Reactor Support Structure, Silo, Guard Vessel, Core-unit, and other structures, systems and components.

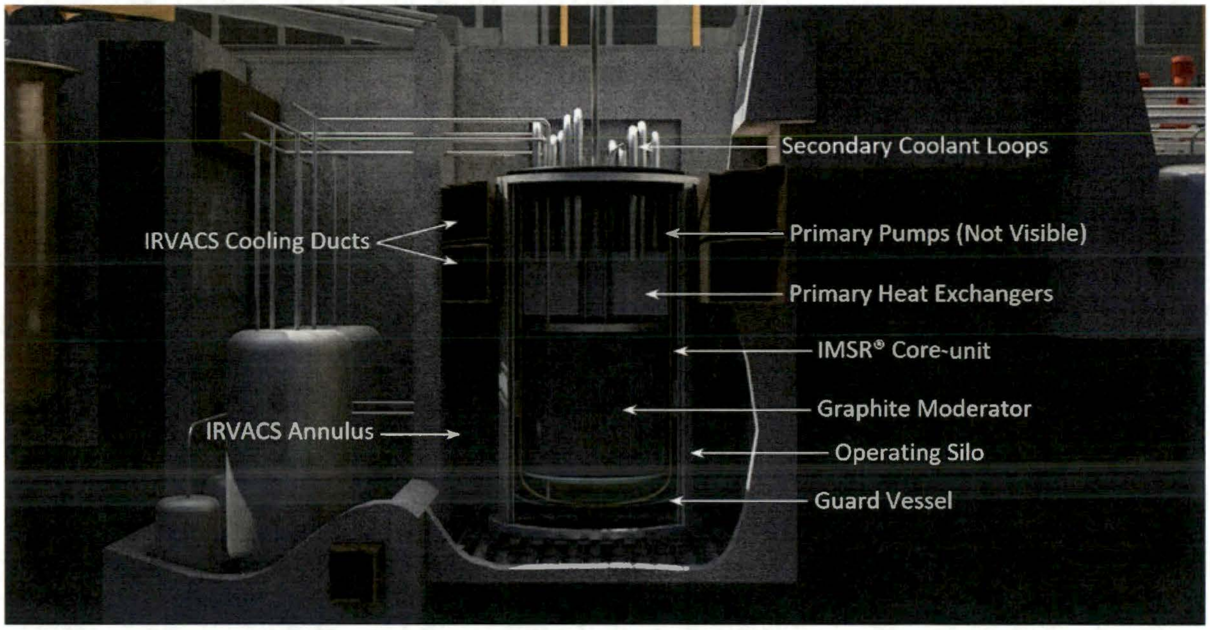
The IRVACS [ ].

Irradiated Fuel AIR Cooling System (IFACS)

The IFACS is a key safety system for the IMSR®. The IFACS functions as a [ ] not subject to the traditional failure modes of emergency heat removal systems such as loss of power. The IFACS is designed to remove heat so that the irradiated fuel salt remains [ ]. The IFACS is capable of removing the [ ].

The IFACS [ ].

Figure 4 The IMSR® Core-unit, Guard Vessel, Reactor Support Structure, and IRVACS ducting in an Operating Silo. Also shown is the Secondary Coolant system piping configuration and interface to the Core-unit as well as the IRVACS annulus surrounding the Silo.



Secondary Coolant System

The purpose of the Secondary Coolant System is to deliver heat from the Primary Heat Exchanger to the Secondary Heat Exchanger, where the heat is transferred to the Tertiary Salt Loop. Figure 4 shows the relationship of Secondary Coolant System piping to the Core-unit and other structures, systems and components. [ ].

The Secondary Coolant System [ ].

Cover Gas & Off Gas Management System

[

] over its 7-year operational life. A separate portion of the Cover Gas System provides [

].

The Cover Gas System [

]. During critical power operation, the Cover Gas System [

].

The Core-unit [

].

].

Cover Gas and Off Gas Management System [

].

Makeup Fuel System (MFS)

The purpose of the MFS is to provide the initial fuel load for new Core-units and to periodically add fuel to the reactor during operation to maintain the reactivity of the core and maintain the fuel temperature in the core at the desired value.

Initial fuel load is "start-up" fuel; fuel added during operations is "make-up" fuel. The system has a safety function to limit the rate and amount of reactivity that can be added to the core to ensure the fuel temperature does not increase in an uncontrolled manner. The system also ensures that fuel outside the Core-unit cannot go critical and meets regulations for safeguards.

The system operates intermittently, is normally isolated from the Core-unit, and kept at, or near, atmospheric pressure. [

].

The MFS [ ].

Irradiated Fuel System (IrFS)

The primary purpose of the IrFS is to remove the fuel from the Core-unit and transfer the fuel to storage tanks for long term on-site storage. This system [ ]. The system can store all of the irradiated fuel generated over the 60-year life of the plant. At [ ], the system [ ].

Below are the main functions of the Irradiated Fuel System:

- [ ].
- [ ].
- [ ].
- [ ].
- [ ].
- [ ].

The Irradiated Fuel System [ ].

Instrumentation and Control

In general, the control functions are not challenging in terms of complexity and performance due to the passive and inherent safety design features of the IMSR®. The I&C system's main functions deal fundamentally with integrated control of production, interlocks for safety coordination, and monitoring system status. Compared to conventional nuclear technology, some of the in-core instrumentation and process equipment for the salt systems operate in a higher temperature environment, [ ].

The I&C architecture is designed for high reliability and robustness against internal failures and external events to ensure that the control functions are available. The use of [ ] achieves high reliability. The [ ].

Critical equipment is also qualified to ensure credited safety functions are available for common-mode events such as earthquakes or extreme environmental conditions that may be caused by postulated initiating events. The Secondary Control Area (SCA) [ ]. The SCA and Main Control Room (MCR) [ ] are also provided.

The system design employs redundancy in systems performing safety or important power production functions to achieve high reliability and fault tolerance in the system. This approach is most effective if the redundant systems and equipment are independent of each other such that failures do not propagate to affect the backup system/equipment, nor do common-mode events (e.g., fire) cause failure of the redundant system/equipment at the same time.

For the IMSR® I&C, [ ]. The design of the I&C architecture ensures [ ].

There are [ ].

V Core-unit Description

The critical feature of IMSR400 innovation is the integration of the primary reactor components into a sealed and replaceable reactor vessel called the Core-unit. The replaceable IMSR400 Core-unit ensures that the materials' lifetime requirements of all reactor core components are not limiting factors, which has been a challenge for the immediate commercialization of MSRs.

The Core-unit is comprised of the following items:

- Reactor Vessel,
- Fuel Salt,
- Primary Pumps,
- Graphite Moderator,
- Shutdown Rods,
- Primary Heat Exchangers, and
- Connections to other systems.

[

]. Piping connections associated with the Core-unit are provided for the:

- Secondary Coolant System,
- Cover Gas & Off Gas Management System,
- Fuel System, and
- Irradiated Fuel System.

Reactor Vessel

The Reactor Vessel is an upright, [ ] cylinder. It contains the full inventory of liquid fuel salt and there are no external fuel salt piping loops associated with the Reactor Vessel. All the nuclear heat fission energy is generated within the Reactor Vessel. [

], the Reactor Vessel forms the primary, nuclear boundary during normal operation, anticipated events, and Design-Basis-Accidents (DBAs).

The Reactor Vessel boundary performs the following functions:

- Contains the fuel salt,
- Provides a flow circulation path for the fuel salt, and
- Provides a support (anchor point) for the core internals.

[

].

The Reactor Vessel is monitored and inspected [

]. The Reactor Vessel also sees significant neutron flux. The flux must be below the alloy embrittlement limit, and the limit [

].

The Reactor Vessel itself is a passive boundary. However, instrumentation is used in the Reactor Vessel to measure:

- Temperature,
- Pressure,
- Neutron flux, and
- Fuel salt level.

The Reactor Vessel [

]. Due to these factors, preliminary safety analysis has demonstrated that vessel failure does not occur in any DBA.

A Guard Vessel surrounds the Reactor Vessel in the event of a Beyond Design Basis failure of the vessel, to catch and contain any leaked fuel salt. The Guard Vessel, however, [
].

The Reactor Vessel is designed to not require maintenance over its 7-year nominal lifetime.

Liquid Fuel Salt

The IMSR® operates by fissions of low-assay low enriched uranium (LEU) [ ] dissolved in a molten primary coolant comprised of a fluoride salt-mixture. The primary purpose of the fuel salt-mixture is delivery of the low-enriched fissile uranium into the IMSR® graphite core for heat generation through a sustained fission chain reaction and subsequent transportation of the heated salt to the Primary Heat Exchangers. The [

] over the 7-year lifetime. The lower melting temperature of the fuel salt-mixture relative to the operational temperature range implies that the fuel salt-mixture will be molten during normal operation ensuring uniform distribution of the fuel and fission products. Fluoride fuel salt-mixtures offer high potential for nuclear applications as they generally have the following essential characteristics:

a) High boiling temperatures.
b) Low vapor pressures.
c) High heat capacities.
d) Low chemical reactivity; and
e) High solubility of fission products.

In any potential emergency involving a sudden temperature increase, the core negative temperature reactivity coefficient will inherently stabilize the reactor such that the IRVACS can passively remove the heat it produces.

The uranium fuel in the form of uranium [ ], is dissolved in a eutectic mixture of [

]. The fuel salt is thus an integral system – nuclear fuel, coolant, and heat transfer medium. An integral system provides the basis for a less complicated reactor configuration and enhanced positive safety attributes. Using a liquid fuel eliminates the need for fuel cladding operating under high pressure and in a highly radioactive environment. The fuel salt is impervious to radiation and maintains a homogeneous mixture of the fuel and coolant.

Primary Pumping System

The Primary Pumping System performs the essential function to circulate the fuel salt through the Core-unit. Its purpose is to provide enough flow through the Primary Heat Exchangers and Moderator to facilitate full power operation without exceeding the material temperature limits of the Core-unit components.

The system [

] system. It directly [

], Containment, and [

[ ].

The [ ] is part of the [ ]. As a result, the design has:

- high integrity and leak-tightness,
- high temperature and radiation resistance,
- ability to accommodate thermal expansion, and
- monitoring provisions to detect leaks.

The primary pumping system is wholly contained within the sealed Core-unit.

Graphite Moderator

The purpose of the Graphite Moderator is to provide the medium for slowing down neutrons to promote the nuclear chain reaction. The Graphite Moderator core design [

], to the Primary Heat Exchangers.

The graphite [

].

Shutdown Rods

IMSR® reactor shutdown (i.e., sub-criticality) is not required to reach a safe end-state for any Anticipated Operation Occurrence (AOO) or DBA. The safe end-state for IMSR® is defined to be the reactor at low power, the reactor vessel temperature within acceptable limits, and no fuel boiling. However, as a defense-in-depth safety measure, and for operational purposes, the IMSR® design includes the Shutdown Mechanism (SDM) as an independent means of shutting down the reactor.

The purpose of the Shutdown Mechanism (SDM) is to bring the reactor to a sub-critical state. The SDM makes use of Shutdown Rods to bring the reactor to a shutdown sub-critical state, which would eventually result in cooldown to a cold condition as decay heat subsides. The safe shutdown state for the IMSR® will employ the Canadian definition of a guaranteed shutdown state. The guaranteed shutdown state is defined as a reactor state with sufficient negative reactivity to ensure subcriticality in the event of any process failure and for which administrative safeguards are in place to prevent net removal of negative reactivity. After the reactor is shutdown [

].

[

].

The Shutdown Rods [ ]. When power is lost, the rods drop under the force of gravity. Under normal operation, the I&C systems [

], eventually resulting in cooldown to a cold condition as decay heat subside.

Primary Heat Exchangers (PHX)

The PHXs provide heat transfer between the circulating fuel salt and a separate closed-loop secondary coolant salt. The PHXs [ ]. The PHXs [ ] and transfers the heat to the secondary coolant salt and then [ ] the fuel salt [ ]. The fuel salt then [ ].

The secondary coolant salt [ ]. This coolant salt transfers the heat away from the reactor core while being isolated from the highly radioactive primary fuel salt liquid. The secondary coolant salt [

].

The PHXs design transfers the total core heat load, which is equal to the thermal power produced in the reactor core, plus the additional heat load from DBAs, including overpower events.

[ ]. The heat exchanger tube size and arrangement [

].

Connections to Other Systems

Other reactor piping systems connected to the Core-unit include the:

- Makeup Fuel System,
- Secondary Coolant System, and
- Off Gas Management System.

VI Core-unit Operations

The IMSR® Core-unit is sealed during operation. The sealed Core-unit concept has both safety and economic advantages. This configuration restricts even minute amounts of volatile fission products from escaping to the environment. The Core-unit is replaced every 7 years [ ]. The IMSR400 plant design sustains eight Core-unit replacement cycles giving it a 56-year operational lifetime. At the end of each 7-year cycle, the fuel salt is discharged to storage in containment, and the used Core-unit is similarly placed in a storage silo within the plant. The stored Core-unit remains in storage for the remaining plant life. The IMSR® power plant design incorporates two Core-unit Silos; this accommodates switching to a new Core-unit every seven years. One Silo is for the operating Core-unit, and one Silo is for storage of either a standby Core-unit or a spent Core-unit, depending on the life-cycle stage of the plant. This is explained in the following table:

Life-cycle Stage	Silo A	Silo B
Years 1 through ~7	Operating IMSR® Core-unit	Standby (unfueled) IMSR® Core-unit
Years 8 through ~14	Defueled IMSR® Core-unit cools and radioactively decays	Operating IMSR® Core-unit
Years ~15 through 21	At the beginning of year ~15, the defueled IMSR® Core-unit is moved to storage. A new IMSR® Core-unit is inserted and begins operation.	Defueled IMSR® Core-unit cools and radioactively decays

This process of alternating between operating and storage/standby continues through the plant life which is planned to be 8 cycles or 56 years.

Primary means of cooling and decay heat

In the IMSR® design, the primary means of reactor cooling occurs when the fuel salt flows convectively, and its heat is transferred out through the PHXs under normal conditions. The Core-unit is also passively cooled by the IRVACS, [ ] that transfers heat to the atmosphere. The IRVACS is continuously in operation and is sized to remove decay-heat, including after an accident where the normal heat removal means might not be available.

Low pressure and high temperature operation

The IMSR® operates at near-atmospheric pressure rather than 70-160 atmospheres of pressure, as is the case with conventional nuclear. Further, the IMSR® design removes the possibility of a system overpressure condition resulting from chemical reactions since all materials inside the IMSR® Core-unit have high intrinsic chemical compatibility. Furthermore, there is very little stored energy in the primary loop (i.e., no mechanism for the generation of high-enthalpy steam), during and after transients, or system upsets. Therefore, the IMSR® does not require expensive high-pressure reactor vessels, high-pressure containment, or active safety systems.

Reactor physics and reactivity control

There are three ways to control reactivity when the IMSR® is critical. These are 1) the negative temperature reactivity coefficient, 2) make-up or fresh fuel addition, and 3) the SDM to terminate the fission reaction and shut down the reactor.

The IMSR® controls reactor power inherently without the automatic manipulation of a reactivity control device. The inherent feature that controls reactivity in the short term is the highly responsive negative core temperature coefficient of reactivity.

Negative temperature reactivity coefficient

The IMSR® design has a strong negative reactivity coefficient of temperature. This inherent feature provides a self-governing and stable temperature regime that establishes the inherently safe operating profile of the IMSR®. The rapid response characteristic and [ ] also allow for load-following capability, which enables an IMSR® to back up variable wind and solar power generation. Along with this fast-acting stability, is long term stability for load-following [ ].

Furthermore, the IMSR® design ensures that in an accident which increases reactor power (e.g., fueling error) or fuel salt temperature (e.g., loss of normal heat removal), the core negative temperature reactivity coefficient inherently stabilizes the reactor such that the IRVACS can passively remove the heat it produces.

Makeup fuel addition, reactivity control, and reactor shutdown

Both start-up and makeup fuel are [ ], the enrichment for the initial fuel load and makeup fuel [ ].

A typical operation would be [ ].

Periodic makeup fuel additions over Core-unit life accounts for reactivity changes due to fuel burnup. This is analogous to rod withdrawal in a light water reactor. Short term power transients are controlled by the inherent response of the negative temperature/power reactivity coefficient of the design.

IMSR® reactor shutdown [ ], as discussed earlier. The safe end-state for IMSR® is defined above. However, as a defense-in-depth safety measure, and for operational purposes, the IMSR® design includes the Shutdown Mechanism (SDM) as an independent means of shutting down the reactor.

VII Regulatory Requirements and Related Guidance

Regulatory Requirements for Principal Design Criteria

The General Design Criteria (GDC) in 10 CFR 50, Appendix A provide the minimum requirements for the principal design criteria (PDC) for water-cooled NPPs. The PDC establish the necessary design, fabrication, construction, testing, and performance requirements for SSCs important to safety. The language of 10 CFR 50 Appendix A goes further to assert that the SSCs important to safety are those that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

The requirements for the contents of an application for a construction permit specified in 10 CFR 50.34 (a) (3) include submittal of the PDC for the facility. It goes further to describe the information as the design basis and information relative to materials of construction, general arrangements, and approximate dimensions sufficient to provide reasonable assurance that the design will provide an adequate margin for safety. TEUSA is not currently planning to pursue an application for construction, either under Parts 50 or 52, however, the contents of application requirements for an SDA application are governed by the requirements contained in 10 CFR 52.137 (a)(3). Upon close inspection, the requirements for submittal of PDC in an application for an SDA are identical to the requirements presented in 10 CFR 50.34 (a)(3).

Relevant Regulatory Guidance for Developing Principal Design Criteria

In an effort to support the development of new non-light water reactor designs, the U.S. NRC and U.S. Department of Energy (DOE) implemented a joint initiative to assess the GDC and determine the extent to which they apply to non-LWR designs and to propose amended or additional criteria that address non-LWR design features. The joint initiative resulted in the recent publication of Regulatory Guide 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors" in April 2018 (Reference 4).

Regulatory Guide (RG) 1.232 provides a set of advanced reactor design criteria (ARDC) that serve the same purpose for non-LWRs as the GDC serve for LWRs. In developing the ARDCs, the NRC considered different technologies including sodium fast reactors, lead-cooled fast reactors, gas-cooled fast reactors, modular high-temperature gas reactors, fluoride high-temperature reactors, and molten salt reactors. As developed, the ARDCs are intended to be generally inclusive for all the designs mentioned but the staff recognizes that it would be difficult to be completely inclusive given the variety in technologies and designs. The RG goes further and provides technology-specific non-LWR criteria for a sodium fast reactor (SFR-DC) and a modular high-temperature gas reactor (MHTGR-DC). As presented in the RG, the NRC intended that the ARDC apply to the technology types included in the DOE report; however, the NRC acknowledges that one or more of the criteria from the SFR-DC or the MHTGR-DC may be more applicable to other technologies being evaluated.

Relevant excerpts from the guidance for developing PDC using RG 1.232 are provided:

- "Applicants may use this RG to develop all or part of the PDC and are free to choose among the ARDC, SFR-DC, or MHTGR-DC to develop each PDC after considering the underlying safety basis for the criterion and evaluating the rationale for the adaptation described in this RG."

- "(A) non-LWR applicant would not need to request an exemption from the GDC in 10 CFR Part 50 when proposing PDC for a specific design."
- "Another example is the molten salt reactors (MSR) that use liquid fuel. An MSR designer may need to develop new PDC for liquid fuel and systems to support this design."
- "In each case, it is the responsibility of the designer or applicant to provide not only the PDC for the design but also supporting information that justifies to the NRC how the design meets the PDC submitted, and how the PDC demonstrate adequate assurance of safety."

The PDC in this report are provided for use by future license applicants seeking to license or construct the TEUSA IMSR® design and are intended to satisfy the regulatory requirements for principal design criteria in the license applications mentioned above.

VIII Process for Developing Principal Design Criteria

This section discusses the process used by TEUSA to develop the principal design criteria (PDC) for the IMSR® design. The initial starting point for the assessment process was the set of ARDC listed in RG 1.232, Appendices A, B, and C (Reference 4).

After consideration of the design criteria contained in the Appendices of RG 1.232, TEUSA concluded that the ARDC that most closely represents the IMSR® technology are those contained in Appendix B (SFR-DC) of the RG. For that reason, the SFR-DC were used for the PDC assessment process for an IMSR® facility.

Since the SFR-DC are written to be inclusive of different types of sodium reactors, and the IMSR® is a fluoride molten salt reactor design, there are certain design criteria that are not applicable to the IMSR®. An example of such a design criterion would be one that provides design requirements for an emergency core cooling system. The IMSR® design does not employ a separate emergency core cooling system as part of its design. In cases when the regulatory guide ARDC relate to SSCs that are not included in the IMSR® design, the specific criteria are considered not applicable and will be excluded from the set of PDC for the IMSR®. The TEUSA basis and justification for each PDC are presented as part of the assessment process.

IX Principal Design Criteria Assessment for the IMSR®

This section presents the TEUSA assessment of selected advanced reactor design criteria (ARDC) to the key features and design of the IMSR® facility. TEUSA will use the language of the ARDC as its PDC when the language of the ARDC can be directly applied to the IMSR® design. In instances where the ARDC are modified to reflect the IMSR® design, the exact language of proposed PDC will be presented followed by justification for the modified PDC.

The IMSR® design is a liquid fueled molten salt cooled reactor [ ]. TEUSA has chosen to use the reference set of SFR-DC contained in Appendix B to RG 1.232 (Reference 4) as the initial starting point for developing its PDC. For ease of reference and discussion, the numbering in the TEUSA table below follows the design criteria numbering contained in Appendix B of RG 1.232. For example, SFR-DC–1 will be presented as TEUSA-1 in the table.

The results are presented in tabular form for each PDC and are shown sequentially below.

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA-1: Quality Standards and Records	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[

	]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA-2: Protection Against Natural Phenomena	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
	[ ]
	[ ]
	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA-4: Dynamic and Environmental Effects Design Basis	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[

	]

Title of Principal Design Criteria	Language of Principal Design Criteria
ARDC 6-9 (reserved)	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA-10: Reactor Design	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[

	]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA-11: Reactor Inherent Protection	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA-12: Suppression of Reactor Power Oscillations	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA-13: Instrumentation and Control	[ ]
	[ ]
TEUSA Supporting Basis for Assessment	[

	]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA-14: Primary Coolant Boundary	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–15: Primary coolant system design	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–16: Containment Design	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[

	]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–17: Electric Power Systems	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[

	]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–18: Inspection and Testing of Electric Power Systems	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–19: Control room	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

	]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–20: Protection System Functions	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[

	]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA-21: Protection System Reliability and Testability	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA-22: Protection System Independence	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA-23: Protection System Failure Modes	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA-24: Separation of Protection and Control Systems	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[

	]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–25: Protection System Requirements for Reactivity Control Malfunctions	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[

	]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA-26: Reactivity Control Systems	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[

	]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA-27: Combined Reactivity Control Systems Capability	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA-28: Reactivity Limits	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–29: Protection Against Anticipated Operational Occurrences	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–30: Quality of primary coolant boundary	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA-31: Fracture Prevention of Primary Coolant Boundary	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–32: Inspection of Primary Coolant Boundary	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA - 33: Primary Coolant Inventory Maintenance	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA-34: Residual Heat Removal	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA-35: Emergency Core Cooling	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–36: Inspection of Emergency Core Cooling System	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–37: Testing of Emergency Core Cooling System	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–38: Containment Heat Removal	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–39: Inspection of containment heat removal system	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–40: Testing of Containment Heat Removal System	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA – 41: Containment atmosphere cleanup	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–42: Inspection of containment atmosphere cleanup systems	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–43: Testing of Containment Atmosphere Cleanup Systems	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–44: Structural and Equipment Cooling	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–45: Inspection of Structural and Equipment Cooling Systems	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–46: Testing of Structural and Equipment Cooling systems	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–50: Containment Design Basis	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[

	]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–51: Fracture Prevention of Containment Pressure Boundary	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–52: Capability of Containment Leakage Rate Testing	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–53: Provisions for Containment Testing and Inspection	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–54: Piping Systems Penetrating Containment	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–55: Primary Coolant Boundary Penetrating Containment	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–56: Containment Isolation	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–57: Closed System Isolation Valves	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–60: Control of Releases of Radioactive Materials to the Environment	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–61: Fuel storage and Handling and Radioactivity Control	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–62: Prevention of Criticality in Fuel Storage and Handling	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–63: Monitoring Fuel and Waste Storage	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–64: Monitoring Radioactivity Releases	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–70: Intermediate Coolant System	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–71: Primary Coolant Purity Control	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–72: Salt Fluid Heating Systems	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–73: Sodium Leakage Detection and Reaction Prevention and Mitigation	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–74: Sodium/Water Reaction Prevention/Mitigation	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–75: Quality of Intermediate Coolant Boundary	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–76 – Fracture Prevention of the Intermediate Coolant Boundary	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–77: Inspection of the Intermediate Coolant Boundary	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA–78: Primary Coolant System Interfaces	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[ ]

Title of Principal Design Criteria	Language of Principal Design Criteria
TEUSA– 79: Cover Gas Inventory Maintenance	[ ]
TEUSA Assessment	[ ]
TEUSA Supporting Basis for Assessment	[

	]

X IMSR® Principal Design Criteria Summary

This section summarizes the principal design criteria assessment for the IMSR® design. For reasons presented earlier in this paper, TEUSA developed its principal design criteria using the SFR-DCs listed in Appendix B of RG 1.232 for a sodium cooled fast reactor. TEUSA assessed the draft SFR-DC against the SSCs in the IMSR® design. The TEUSA specific IMSR® PDCs were either SFR-DCs adopted without modification or SFR-DCs modified to account for the unique features of the IMSR® design. In some cases, several SFR-DCs were not adopted.

The assessment process determined that forty-four SFR-DCs could be directly adopted for the IMSR® design without modification. Ten SFR-DCs could be adopted with some modification of the draft criteria to reflect the SSCs incorporated into the IMSR® design or the planned operational characteristics of the IMSR® design. Ten SFR-DCs were not adopted because the draft criteria are not relevant for the IMSR® design.

TEUSA adopts the set of 54 principal design criteria for the IMSR® design presented in Section IX above with the proposed modifications necessary for the IMSR® design. The principal design criteria explicitly excluded from the set of IMSR® principal design criteria are repeated below for ease of reference. These criteria were determined to be not relevant to the SSCs that are incorporated into the IMSR® design.

The excluded design criteria are:

Excluded Criterion	Reason for Exclusion
SFR-DC-21 Protection System Reliability and Testability	[ ]
SFR-DC-22 Protection System Independence	[ ]

Excluded Criterion	Reason for Exclusion
SFR-DC-24 Separation of Protection and Control Systems	[ ]
SFR-DC-33: Primary Coolant Inventory Maintenance	[ ]
SFR-DC-35: Emergency Core Cooling	[ ]
SFR-DC-36: Inspection of the Emergency Core Cooling System	[ ]
SFR-DC-37: Testing of the Emergency Core Cooling System	[ ]

Excluded Criterion	Reason for Exclusion
SFR-DC-55: Primary Coolant Boundary Penetrating Containment	[ ]
SFR-DC-73: Sodium Leakage Detection and Reaction Prevention and Mitigation	[ ]
SFR-DC-74: Sodium/Water Reaction Prevention/Mitigation	[ ]

Abbreviations & Acronyms

AOO – Anticipated Operational Occurrence

ARE – Aircraft Reactor Experiment

BeF_2 – Beryllium Fluoride

CFR – Code of Federal Regulations

CNSC – Canadian Nuclear Safety Commission

Cs - Cesium

DBA – Design Basis Accident

DMSR – Denatured Molten Salt Reactor

FSST- Fuel Salt Storage Tank

I&C – Instrumentation and Control

IFACS – Irradiated Fuel Air Cooling System

I-NPP – IMSR Nuclear Power Plant

IMSR® – Integral Molten Salt Reactor

IRVACS – Internal Reactor Vessel Auxiliary Cooling System

KF – Potassium Fluoride

LIF – Lithium Fluoride

MCR – Main Control Room

MFS – Makeup Fuel System

MW – Megawatt

MWe – Megawatt Electric

MWth – Megawatt Thermal

MSR – Molten Salt Reactor

MSRE – Molten Salt Reactor Experiment

NaF – Sodium Fluoride

ORNL – Oak Ridge National Laboratory

PDC – Principal Design Criteria

PHX – Primary Heat Exchanger

PSA – Probabilistic Safety Assessment

R&D – Research and Development

RAB – Reactor Auxiliary Building

REP – Regulatory Engagement Plan

SCA – Secondary Control Area

SDA – Standard Design Approval

SDM – Shutdown Mechanism

Sr - Strontium

SS – Stainless Steel

TEI – Terrestrial Energy, Inc.

TEUSA – Terrestrial Energy USA, Inc.

U.S. – United States

VDR – Vendor Design Review

Xe - Xenon

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