TERRESTRIAL
ENERGY USA

IMSR® Core-unit Definition

Applicable Structures, Systems and Components

Revision 1, May 2022

Abstract

TEUSA's long term licensing objective for the IMSR® design is to obtain a Standard Design Approval (SDA) for the IMSR® Core-unit. An important component of a 10 CFR Part 52 SDA application for the IMSR® Core-unit is identification and description of the structures, systems, and components (SSCs) of the IMSR® Core-unit. This whitepaper provides a general overview description of the IMSR® design and a more detailed description of IMSR® Core-unit SSCs.

Non-Proprietary

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I. Purpose

The purpose of this white paper is to provide information about the IMSR400 design and to define the important systems and structures that comprise the Integral Molten Salt Reactor (IMSR®) Core-unit.

This white paper also supports the identification of interfaces, boundary conditions, and requirements necessary for the content of an application for a Standard Design Approval (SDA) of the Core-unit under 10 CFR Part 52, Subpart E.

Interface requirements are those requirements related to the interface and boundary conditions associated with the Core-unit. The interfaces will stem from the dependency of the structures, systems, and components (SSCs) that are within the scope of the application for a Core-unit SDA, as well as on the functional and operational characteristics of SSCs that are not within the scope of the SDA.

Interfaces and boundary conditions can be distinct; however, the terms can also be used interchangeably. Nonetheless, together, they describe the limitations, constraints, assumptions, and conditions to define the relationship between the Core-unit and the remainder of the power plant.

An interface could include a programmatic requirement or an operational assumption about system performance of the Core-unit. Whereas a boundary condition could be a physical constraint or an explicit limit on an interfacing system or component, or a similar restraint or limitation associated directly with the Core-unit. Additionally, a boundary condition may be a well-defined physical point of separation, or departure, between an interfacing system and the Core-unit. Additional information specific to interface requirements will be provided at a later date once the IMSR400 design is completed in order to support the application process.

The information requirements for a Core-unit SDA application are 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.

This document provides a general overview description of the IMSR® design. It provides a description of IMSR® Core-unit SSCs to support meeting the information requirements in 10 CFR 52.137 related to identifying, defining, and describing the IMSR® Core-unit design basis in an SDA application for the Core-unit. This paper also provides an overview of the main plant buildings and SSCs that make up a dual unit reactor IMSR400 Nuclear PowerPlant (I-NPP).

II. Introduction

Terrestrial Energy USA, Inc. (TEUSA) is developing the IMSR® design to provide electricity, or process heat, or both to U.S. industry. TEUSA is planning for the first commercial deployment of this technology within 10 years. The IMSR® is a Generation IV advanced reactor that employs a fluoride molten salt reactor (MSR) design. The reference IMSR® nuclear power plant (I-NPP), consists of two nuclear islands that produce a total of 884 MWth, (442 MWth per Core-unit) for about 390 MWe ((195 Mwe per steam turbine) of net electric output. This plant configuration is herein referred to as the IMSR400. The IMSR400 also has the potential to export 600 °C of heat for industrial applications, or some combination of both. The I-NPP includes an adjacent steam plant and turbine buildings for each nuclear island that contains non-nuclear-grade, industry-standard power 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 2) 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 as well as the status of those review activities. 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 an SDA of the IMSR® Core-unit. This white paper is one in a series of technical documents that support the TEUSA SDA application development efforts.

Company Background

TEUSA performs licensing activities for IMSR® design regulatory applications in the U.S. but does [ ] for the IMSR® design. 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 Charlotte, NC and Greenwich, CT.

Canadian Nexus

TEUSA [ ]. TEUSA leverages the ongoing engineering and regulatory work that TEI accomplishes as TEI 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 to support the regulatory reviews in both countries 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 additional clarification related to the corporate structure for both TEUSA and TEI.

Figure 1. TEUSA-TEI Corporate Relationship

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Licensing Strategy and Objective

The REP outlines the regulatory strategy for TEUSA licensing activities in the U.S. The licensing strategy of TEUSA is to incrementally advance the IMSR® licensing efforts in a purposefully planned and financially informed fashion to support a commercial operation date for the first U.S. plant in the next 10 years. 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 and requesting feedback, TEUSA can receive specific information regarding those areas that may require further testing or technical analyses. Additionally, the NRC will be better able to estimate the resource and schedule requirements necessary to conduct regulatory activities associated with IMSR® licensing.

TEUSA's 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 systems that perform 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 arguments supporting TEUSA's long term licensing strategy for seeking an SDA for the Core-unit portion of the IMSR® include:

- Demonstrating incremental licensing progress resulting in a reduction of overall licensing risk,
- Reducing initial development costs by deferring portions of the plant review to subsequent licensing steps such as during a construction permit or combined license application, and
- Supporting future customer options such as new plant deployment.

This white paper provides a general overview description of the IMSR® design and a more detailed description of IMSR® Core-unit SSCs that form a 'major portion' of a planned SDA application for the IMSR® Core-unit. The information requirements for an SDA application for the Core-unit is a subset of the information requirements supporting an application for a construction permit or combined license, thereby supporting the longer-term goal of IMSR® deployment in the next 10 years. It is also important to note that pursuit of an SDA in no way hinders the development of a construction permit application should a future applicant determine that licensing under Part 50 is a desired approach.

If more details about TEUSA's licensing activities and objectives are needed, please refer to the REP previously provided to the NRC (Reference 2).

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/coolant, 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 IMSR400 design achieves excellence in nuclear safety with the intrinsic properties of the design. Although both conventional reactors and the IMSR400 use Low Enriched Uranium (LEU), the IMSR400 uses LEU in a liquid form, not a solid form. The IMSR400 uranium fuel in the form of uranium tetrafluoride (UF4), is dissolved in a salt mixture of low-cost [ ]. The benefit of this mixture is the minimization of tritium production. The molten salt (fuel salt) is an integral working fluid – nuclear fuel, coolant and heat transfer medium – providing the basis for a less complex reactor configuration and many safety attributes.

The baseline IMSR400 site would consist of five primary structures:

- Two seismically qualified Reactor Auxiliary Buildings (RABs) containing the Reactor Core-units and associated nuclear and support systems to transfer heat to the steam generating equipment. The RABs are located within the Protected Area.
- Two Turbine Buildings (TBs) containing non-nuclear-grade, industry-standard power equipment, each connected to a Steam Generation Building (SGB) containing pre- heaters, reheaters, evaporators and superheaters. This is located outside of the Protected Area to allow improved access for maintenance; and
- One main Control Building (CB) is shared between the two units and is sited between the two RABs within the protected area. It contains a shared Main Control Room, the major electrical systems, and non-nuclear support systems for the overall facility.

The IMSR400 reactor core operates at a peak average fuel salt temperature of [ ]. This is significantly higher than the melting point temperature of the fuel salt. It has robust coolant properties and intrinsically high radionuclide retention capabilities. It utilizes LEU of less than 5% ^{235}U.

Fission heat is transferred from the reactor core by a secondary coolant salt, also a molten fluoride, which flows in a closed loop through secondary heat exchangers external to the Core-unit. Heat is then removed from the secondary heat exchangers by an inexpensive and common molten tertiary coolant salt, which is pumped from the nuclear area to the steam plant and turbine systems for electricity generation. This method gives the IMSR® the versatility to be a high-temperature industrial heat source that can provide up to 390 MWe of net electricity.

The power plant described in this white paper includes a description of the reactor and power conversion process for creating heat in the reactor core and subsequently transferring the heat to produce electricity. Also included below in Figure 3 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.

Single failure analysis is done system-by-system, as applicable. However, the single-failure criterion may be used for active systems for operational reliability considerations. For the passive systems, specifically

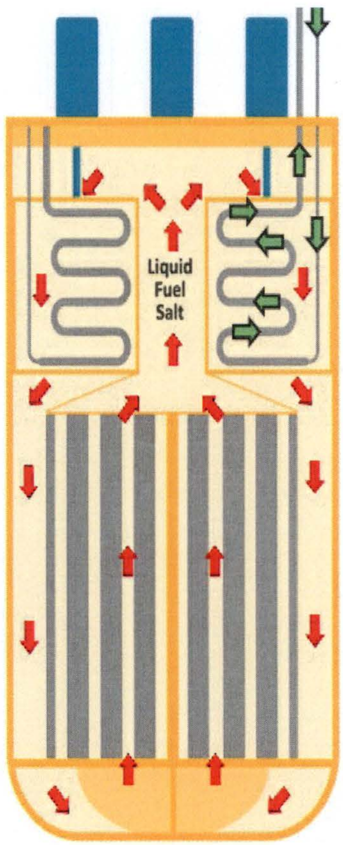
the boundaries of the Internal Reactor Vessel Auxiliary Cooling System (IRVACS) and of the containment, the SSCs will be designed, manufactured, constructed, inspected, and maintained to preclude consideration of failure of function (such a failure that potential leads to a large release will be practically eliminated); and the safety analysis / Probabilistic Risk Assessment (PRA) will show that Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs) and representative Beyond Design Basis Accidents (BDBAs) will not cause boundary failure. Passive components may be exempt from the single-failure criterion for those components that are designed and manufactured to high standards of quality, that are adequately inspected and maintained in service, and that remain unaffected by any other systems Postulated Initiated Events (PIEs) in the affected system.

Reactor

The IMSR® is a liquid-fueled, thermal spectrum, burner-type, fluoride MSR 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 the primary reactor components, including the pumps and heat exchangers, are inside a sealed and replaceable Core-unit with the Reactor Vessel (RV) 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 RV. 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 enters the graphite moderator core region near the bottom of the RV. As it passes upward through flow channels in the graphite, thermal neutrons cause fission of the ^{235}U contained in the fuel salt. The fissioning of the fuel raises the temperature of the fuel salt as it passes through the moderator region. The fuel salt, [ ], exits the graphite moderator, passing from the core region of the reactor, where it enters the Primary Pumps and Primary Heat Exchangers (PHX). Upon exiting the PHXs, the fuel salt flows downward to the bottom of the RV where it begins the cycle again. A simplified schematic showing the flow sequence of IMSR® fuel salt is shown in Figure 2 below.

Figure 2: Simplified Schematic of IMSR® Fuel Salt Flow



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 IMSR400 site includes 2 reactor units and has a small footprint (about 12 hectares or 30 acres) and a small security perimeter (approximately 220m x 160m).

Each site includes two Reactor Auxiliary Buildings, two Steam Generation Buildings, two Turbine Buildings, and a shared Control Building. Also included are the plant support buildings and structures. These include the Maintenance Buildings, Rad Waste Building, Salt Storage Building, Emergency Mitigating Equipment Building, Main Pump House, Fire Water Pumphouse, Cooling Water Outlet Building, Electrical Switchyard, Main Security Building, and Auxiliary Security Building. A graphic showing a generic proposed site layout of an IMSR® site is provided as Figure 3 below.

Figure 3: Proposed Site Layout

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Reactor Auxiliary Building

The RAB is a reinforced concrete structure, which consists of three floors below grade and two full floors above grade, with a large upper area to provide sufficient space for the [ ] located within it. This building is designed to 1) be seismically qualified, 2) be tornado proof, 3) withstand postulated explosion blasts from railway tracks, and 3) provide protection against large aircraft crashes.

Each RAB houses an IMSR® Core-unit within the reactor vault. There are two reactor vaults provided within each RAB to enable the replacement Core-unit to be installed in the adjacent vault before the first Core-unit is replaced after its 7-year operating lifetime. The used Core-unit will then be lifted into one of the six used Core-unit storage silos, using the large RAB overhead crane. There is a total of six used Core-unit silos provided to enable storage of all the used Core-units needed, during the design life

of the IMSR400 Facility. There is also a central vault, known as the Fuel Salt Storage Tank (FSST) Vault, which is located between the two reactor vaults for placement of the FSST, and off-gas tanks. It is not anticipated these tanks will require replacement during the 60-year design life of the IMSR400 Facility, however a means of accessing them remotely is provided.

The major equipment associated with the following system designs have been considered in the latest RAB Layout:

a. Core-units,
b. Guard Vessels
c. Containment,
d. Reactor Support Structures (RSS),
e. Used Core Storage Silos,
f. Initial Fueling System,
g. Make-up Fueling System,
h. Irradiated Fuel System (IrFS),
i. Secondary Coolant System,
j. Tertiary Coolant System, and
k. Internal Reactor Vessel Auxiliary Cooling System (IRVACS) and other safety support systems, located within the RAB,
l. Electrical Systems and Instrumentation and Control (I&C) Rooms
m. Major RAB HVAC Systems, and
n. Fire Protection Systems

The RAB size, defined by the latest RAB layout developed, is used as the basis for this overall facility layout. Both RAB#1 and RAB#2 are essentially identical in design and size, except the access routes to the interconnected buildings and structures are designed to allow the required access to the Control Building (CB) and each RAB.

Turbine Building

The turbine building (TB) is a conventional structural steel building with appropriate siding provided above ground and supported from a reinforced concrete basement. It consists of one level below grade and two levels above grade to house the turbine generator, condenser and the associated feedwater and steam systems and equipment.

Each TB houses the turbine generator, the associated high pressure and low-pressure feedwater heaters, feedwater pumps and main and auxiliary steam systems required to produce the steam for each turbine. Both TB#1 and TB#2 are essentially the same design with no handling of the major equipment provided within them. Each TB is provided with its own overhead crane, sized to lift the loads associated with maintaining the major equipment of the proposed ~200 MWe turbine generator design. The laydown area and access door are also located at the end of the TB to provide sufficient space for the major equipment to be maintained and transported to and from each TB.

The TB also houses the electrical and control equipment associated with the operation of the Turbine

Generator power and control systems that connect back to the main control room, located in the control building.

The steam plant and the associated buildings have no safety function for the I-NPP and are therefore 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 turbine-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.

Steam Generation Building (SGB)

The SGB is a conventional structural steel building with appropriate siding provided above ground and supported from a reinforced concrete basement. It consists of one (1) level below grade and four (4) levels above grade to house the various steam generating equipment and the associated TCS and steam piping connected to them.

The TCS piping, connected to the SCS heat exchangers, located in each RAB, transfers the heated tertiary coolant salt to the various steam generating vessels located in the SGBs. The TCS piping lines run
[

] SGB. These generate high pressure steam suitable for the turbines located in the each of the adjacent Turbine Buildings and provide a separate steam supply to each Turbine Generator (TG), TG#1 and TG#2.

The layout of the SGB has been developed further, taking into consideration latest size and design information for the horizontal steam generating equipment, and is positioned within each SGB to suit the various interconnections required with the TCS piping and the drain tanks and the turbine steam systems piping.

Control Building

The Control Building is a reinforced concrete structure, which consists of three floors below grade and four full floors above grade. This building is designed to be seismically qualified and tornado proof. The CB is located between the two RAB structures and provides support and services to both RAB units.

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. Tunnels and access routes provide for personnel ingress and egress and, for routing of auxiliary, electrical, instrumentation, and communication conduits between the buildings.

This building is fully protected by automatic fire suppression systems, with the exception of the Common Control Room which is attended on a continuous basis. All electrical rooms and control equipment rooms are protected by gas suppression systems. The rest of the building is protected by a pre-action sprinkler system.

Because of the automatic fire suppression systems, fire areas are specified for a [

]. In addition, the complex is served by a dedicated HVAC system which is separate from the HVAC system that services the rest of the building. This system maintains a small positive pressure with respect to the surrounding to reduce potential for ingress of smoke and contaminations. Two egress paths are provided for Main Control Room (MCR) operators to reach the secondary control area.

The Control Building has instrumentation system interfaces with both IMSR® Core-units. The Control Building also has structural interfaces with both Reactor Auxiliary Buildings that houses the IMSR® Core-unit.

Main Control Room and Secondary Control Areas (MCR and SCAs)

The IMSR400 Control Facilities include the MCR and SCA(s). The MCR is located within the Control Building (CB) and is used for both units of a 2-unit plant. The SCAs (one for each unit), are located adjacent to the RAB as seen in Figure 4. References 18, 19, and 20 provide a more detailed design description for the control facilities.

These Control Facilities are seismically qualified, as is the route between them to ensure safe passage in emergency scenarios. The MCR is planned to be approximately [ ]. The MCR contains [ ].
The paths between the seismically qualified and the non-seismically qualified communications channels are physically separated. The SCA is significantly smaller than the MCR. The dimensions for the SCA are [ ].

The SCA contains [ ]. The communications channels for the console pairs are physically separated. This physical separation ensures that if a seismic event disables one communication channel the other will remain intact with no loss of functionality. The SCA is continuously available. All operator control and monitoring can be assumed from the SCA if the MCR is unavailable. These facilities will have habitability systems, including uninterruptible power and breathing air for emergency scenarios, to ensure that at least one of the Control Facilities will operate during all plant states. Emergency operations lighting is powered by the AC Uninterruptible Power System and is provided in the control rooms and other critical areas to support the monitoring of the reactor by the operator.

Shielding, air purification, and climate control systems will be in place in the Control Facilities. Further details regarding these habitability systems, including storage of food and water and duration of habitability will be determined in subsequent analyses. The Control Facilities design will ensure that background noise will not impair regular conversation, maintain an acceptable temperature and humidity, and refresh the room with uncontaminated outdoor air. The SCA is a separate area which has its own HVAC and power systems. Evacuation routes have been planned for in design of the layout of the CB and RAB.

Figure 4. Locations of SCAs and MCR in RAB

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Details for the safety requirements for the control areas with respect to seismic, environmental, containment boundaries, and separation/diversity considerations can be found in References 18, 19, 20. The control facilities are contained within the RAB and as such, they are seismically qualified. The communications channels leading to and from the control areas are both redundant and designed to function through seismic events. This ensures that the control areas will be able to perform their functions after a design basis earthquake. The use of redundant and physically isolated communications busses minimizes the damage potential of events to the extent that system availability is maintained. Additionally, the separation of power and communication signals prevents a fire in one control facility from disabling another control facility.

Standby Diesel Generator Buildings

These buildings are located within the protected area and house the diesel generators that provide backup electrical power for building facilities and selected process systems in the nuclear island. There is one Diesel Generator building for each reactor unit. It is a stand-alone building detached from the RAB and the CB. Each building has two separate compartments, and each compartment contains a redundant, non-seismically qualified standby diesel generator system. They are located such that underground connections can be made to both RAB units and the CB.

Main Security Building (MSB)

The MSB layout developed consists of a three-floor building, with a basement level and two (2) above ground levels to provide the following key facilities:

- Basement: Central Alarm System (CAS), Uninterruptible Power and Armoury.
- Level 1: Ingress and Egress Routes into Facility, Security Office; and
- Level 2: IT Cabinet Rooms for Security Systems, HVAC and gymnasium/ locker rooms.

This revised size of MSB is now included in the latest Facility and Site layouts developed for the IMSR400 Facility.

The MSB basic layout may require further changes during the detail engineering phase when assessing the Design Basis Threat (DBT) requirements and any other features to be considered, however they are not expected to change the overall size of the MSB, presently specified.

Maintenance Building

This building is located within the protected area of the site and houses the various maintenance facilities required for nuclear island operations and maintenance of the various equipment located within it. It is a single floor steel frame building, sized to handle the largest size of equipment that may need to be serviced within this protected area and has access doors provided at each end of the building. It will contain typical mechanical, electrical, and I&C workshops, welding shops, and storage rooms for spare parts, tools, and supplies. There may be small office rooms provided for control and inspection of the work performed within it.

Rad Waste Storage Building

This building is located within the protected area and is used to prepare and store the Intermediate-level radioactive waste (ILW) and Low-level radioactive waste (LLW) that generates in the IMSR400 Facility before shipment.

Emergency Mitigation Equipment Building

This building is located within the protected area to provide space for the various equipment associated with providing "Black Start Power," access route clearance and fire protection equipment/security vehicles.

IV. Structure, System and Component Descriptions – Outside Core-unit

Silos and Reactor Vaults

There are six silos and two Reactor Vaults included in each RAB. The two Reactor Vaults are for operating Core-units, and six silos are for used Core-unit storage. One of either of the two operational Reactor Vaults houses the operating Core-unit for its 7- year operational life; the second Reactor Vault 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 the Reactor Vault into a used core storage silo. The six storage silos only house spent Core-units that have completed the required radioactivity decay cooldown period. The silos and Reactor Vaults interface with the Reactor Vessel and Reactor Support Structure

Salt Leakage Detection

The main concern in monitoring for salt leakage is to ensure that the highly radioactive fuel salt does not breach the various barriers to the environment and, if it does, ensuring that corrective action can be taken as quickly as possible. The potential paths for fuel salt leakage are into the guard vessel from the Reactor Vessel or into the Secondary Coolant System via the Primary Heat Exchangers, or from EHX1 to IRVACS secondary side coolant. In the latter case, the Secondary Coolant System is normally at higher pressure than the fuel salt and so leakage from fuel-to-coolant would only happen in a situation where this pressure difference could not be maintained. Hence, the secondary coolant is also monitored for leakage to ensure that it remains intact as a second barrier to radioactive releases from the fuel salt and to contain any activation products within the secondary coolant salt (which are expected to be small). Finally, the tertiary coolant salt is also monitored for leakage to ensure that it maintains its integrity as a third barrier to releases via possible leaks in the secondary heat exchangers.

The basic method of leak detection is to [

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For fuel salt leakage detection, [

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Reactor Support Structure

The Reactor Support Structure (RSS) is defined as the steel structure supporting the IMSR® reactor Core-unit, the Reactor Vessel (RV) and the Core-unit containment. See Figure 5 for a visual of the Reactor Support Structure. [

]. The best documents describing the RSS and Containment are IMSR400-21200-DBD-002, IMSR400-21200-DD-002, and IMSR400-21200-ASD-005.

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The RV is supported by the [

]. See Figure 5 for the proposed support configuration.

The RV is supported using [

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Figure 5: Conceptual Drawing of RSS Arrangement Around the Containment

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Containment

The main functions of Containment are to:

1. 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 radioactivity in the plant are the reactor core, off-gas storage, and irradiated fuel systems.
2. Control personnel access into containment to protect plant personnel from radiation. [ ].
3. Minimize leakage to assure that normal operation release limits are met, and AOOs and DBAs will not result in exceeding dose acceptance criteria defined in TEI Design Guides. (Note that the dose acceptance criteria will satisfy NRC regulatory requirements.)

Containment consists of the [ ] that houses radioactive materials. This includes radioactive materials (irradiated Fuel Salt and Off-gases) in the Core-unit, Fuel Salt Storage Tanks (FSSTs), Gas Holding Tanks (GHTs), Hot Cell, and connecting piping. In the event of a leak in any of these systems, Containment provides an acceptable leak-tight boundary during DBA scenarios to limit the release of any radioactive materials to the Reactor Auxiliary Building (RAB) and surrounding environment. Containment must also assist in minimizing releases in the unlikely event of a severe accident (BDBA) to the extent practical."

Due to the lack of stored pressure or chemical energy, even in the event of BDBAs, the IMSR400 containment demands are low. The over-pressurization accidents that are the primary concern with water-cooled reactors are not credible failure modes for the IMSR400. [ ].

The air inside the containment is stagnant, and there are instrumentation and monitoring capabilities to measure system parameters such as airborne radiation, temperature, pressure, and oxygen content, and to promptly detect leaks. [ ], which provides additional defense-in-depth and investment protection for fire prevention and corrosion. Five percent is the minimum oxygen concentration below which combustion is not possible, independent of the concentration of the fuel.

The containment volume will pressurize or depressurize as the reactor heats up or cools down during normal operating maneuvers. The system is designed to accommodate these pressure swings; [ ] over the long term, to minimize any leakage in or out. This [ ] will be qualified as a containment extension but will [ ] during normal operation.

The containment is designed to maintain [ ]. Accidents identified to date do not cause any significant pressurization inside containment. The entire system is designed to withstand [ ]. The envelope follows the strict rules of a containment, i.e., an essentially leak-tight structure with a defined design pressure and a defined maximum leak-rate at that pressure.

The design of the containment and its parts ensures that the release of radioactive material to the environment and public is within the acceptable regulatory limits. The containment structure is designed to meet the applicable requirements for a steel containment. Containment penetrations are designed to the same requirements as the containment structure itself and are [

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The Containment system is designed so that [

]. Means are provided for pressure testing afterwards at a specified pressure [ ] to demonstrate structural integrity.

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Figure 6 below is a sketch of the layout of the Core-unit, containment, and RSS within the Reactor Vault:

Figure 6. Core-unit Containment Within the Reactor Vault

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Lower Containment

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The inner diameter of the Containment enclosure is [ ]. Considering internal Reactor Vault plan view dimensions of [ ] when also considering the RSS components. It is assumed that the Containment enclosure will be [ ] structure.

Upper Containment

[ ].

Figure 7 below is a sketch of the Upper Containment portion:

Figure 7. Proposed Support Configuration

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]. The isolation of the applicable lines outside of Containment may be performed by various methods, whether it be [

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Pipe Journals

Pipe Journals are metallic pipes that [

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The pipe journals will be separated into sections. Inside the Pipe Journal, there will be [

].

The purpose of this separation is to [

].

Figure 8 shows a simplified depiction of the Pipe Journals [

].

Figure 8: Location of Seal Plates Relative to Pipe Journals and Containments

[

]

Hot Cell

The Hot Cell is a controlled room which contains all of the valves, compressors and couplings related to the Irradiated Fuel System (IrFS), Makeup Fuel System (MFS), and Initial Fuel System (InFS). [

], no personnel access is needed for the life of the IMSR400. [

].

Purge and Pressure System

The Pressure Control and Purge System will utilize [

], as required.

The Pressure Control and Purge System of the IMSR400 containment will be interconnected during NO. Prior to startup, the active containment volume will be closed.This volume is then purged via vacuum purging to remove any harmful gases and be made inert. The pressure of the containment volume is brought to a slightly negative pressure (relative to the Reactor Vault).

Due to this negative pressure, and any potential leak into containment, the containment volume will be [

]. The purge system will still be connected to allow for periodic inert gas addition if required throughout operation.

[

]. It should be clarified that depending on where an accident occurs, that volume will be closed off.

[

1.
2.
3.
4.
5.

].

Internal Reactor Vessel Auxiliary Cooling System (IRVACS)

The Internal Reactor Vessel Auxiliary Cooling System (IRVACS) is a Safety System specifically designed to remove decay heat from the Fuel in the Core-unit following an emergency shut down of the reactor and when no power is available. Emergency in this context means a reactor shutdown that is not a normal controlled shutdown and is usually triggered by an event whereby normal shutdown procedures are not available.

Implicit in IRVACS requirements are that no other heat sinks are available. No operator intervention is required, decay heat is removed, and IRVACS is fully passive. There are no controls and no equipment that undergoes a change of state., Also, IRVACS does not rely on any instrumentation during its operation.

IRVACS consists of [

], thereby providing adequate redundancy. In addition, and at present, no single Beyond Design Basis Accident (BDBA) will [

]. Each component is designed and constructed to be rugged and robust, such that, no external occurrence (that is, external to IRVACS) will render IRVACS inoperable.

The Internal Reactor Vessel Auxiliary Cooling System (IRVACS) is continuously operating and also

functions as an additional heat sink for residual heat[1] removal of the fueled Core-unit in accident conditions. Due to the importance of this system, it is designed to ensure that, in the case of complete loss of all power, passive cooling will maintain the pressure boundary integrity of the Core-unit and the fuel salt below its saturation temperature. It is therefore not subject to the traditional failure mode of emergency heat removal systems such as loss of power.

IRVACS functions to remove the decay heat when the [

], as well as all AOOs and some DBAs.

IRVACS is designed to be fully passive and capable of removing decay heat and any residual neutron power from the Fuel in the Core-unit for any length of time with no operator action.

IRVACS is designed to be effective during accidents as may occur when no power is available and other heat sinks are likewise unavailable, such as a Station Blackout (SBO). IRVACS is effective for any single DBA or single BDBA in keeping the Core-unit bulk below the preliminary target temperature [ ]. Figure 9 provides a simple schematic of a single loop of the IRVACS basic design.

IRVACS has [

].

More details into the IRVACS design requirements can be found in IMSR400-22370-DR-001 Rev. 2 - Internal Reactor Vessel Auxiliary Cooling System (IRVACS), IMSR400-22370-SC-001 Rev. 2 - Internal Reactor Vessel Auxiliary Cooling System (IRVACS), and IMSR400-22370-DD-001 Rev 1 (References 21,22, 23).

[1] Residual heat consists mainly of decay heat, although there are scenarios where IRVACS will be required for removal of heat produced via the fission chain reaction, such as when there is a delay between when process heat sinks stop removing heat from the Core-unit and the negative temperature reactivity coefficient of the Core-unit has stopped the fission chain reaction.

Figure 9. Simplified Schematic of a Single IRVACS Loop

[

]

Figure 10 shows the general layout of the internals of the Core-unit as they relate to IRVACS. [ ]. Figure 11 is an isometric view of the PHXs and EHX1s.

Figure 12 is a schematic/flow diagram of IRVACS 1a.

Figure 10. General View of Reactor Vessel Internals

[

]

Figure 11. An isometric view showing the PHXs and EHX1 nestled within the RV

[

]

Figure 12. A Schematic/Flow Diagram showing IRVACS 1a

[

]

Figures 13, 14, and 15 on the following pages show various orientations the IRVACS physical arrangement in the RAB. For further descriptive details of the piping arrangement and the air duct arrangements refer to IMSR400-22370-ASD-004. [

]. However, the intent of this document is, at this stage in basic engineering, to demonstrate the favorable design features and viability of IRVACS.

Figure 13. IRVACS physical arrangement in the RAB

[

]

Figure 14. IRVACS physical arrangement in the RAB

[

]

Figure 15. IRVACS physical arrangement in the RAB

[

]

Equipment in IRVACS

The following is a description, with drawings, of the equipment that makes up IRVACS.

Primary Heat Exchanger (PHX)

[

].

Details of the PHX are shown on Figure 16.

Figure 16. Isometric view of one PHX

[

]

Emergency Heat Exchanger 1 (EHX1)

Figure 17 shows [

].

Figure 17. Shows inclined tube support plates

[

]

Emergency Heat Exchanger 2 (EHX2)

Figure 18 shows some of the dimensions and physical features of EHX2 [

].

Figure 18. Sub figures showing the construction details of EHX2

[

]

Thermal Static Flow Control Device

IRVACS transfers heat from the Fuel Salt in the Core-unit to the atmosphere by means of natural circulation. The Fuel Salt can freeze at about [ ] and the IRVACS Coolant Salt can freeze at [ ]. As the heat sink, i.e., the atmosphere, is [ ]. To stay within the definition of "passive" then a [ ].

The basis of the design is described in the Figures 19 and 20.

As EHX1 and EHX2 are both relying on thermal syphoning in a passive sense, [ ]. Thus, to maintain a stable system, a [ ].

Figure 19. Basic Principle of a Temperature Flow Control Device Based on Thermal Expansion

[

]

Figure 20. Axial application in a pipe for a Thermal Flow control Device

[

]

Flow Restrictor

The Flow Restrictor is located in the [

] Fuel Salt flow. As can be seen, the [
]. During accident conditions the fuel salt flow rate through the [
], which is negligible.

Performance of IRVACS

Figure 21 is a presentation of a compilation/summary of the calculation results for IRVACS, IRVACS 1a and PRACS, and normal operation.

Figure 21. Summary of the calculation results for IRVACS, IRVACS 1a and PRACS, and normal operation.

[

]

Figure 22 shows a time dependent relationship of fuel salt temperature and the RV heat exchange, i.e., heat in and heat out. The heat absorption capacity of the RV is also included. Both cases, [ ] are shown. With [

].

Figure 22. Station Blackout IRVACS

[

]

Figure 23 (presented as Graph 3b) shows a time dependent relationship of fuel salt temperature and the RV heat exchange, i.e., heat in and heat out absent the insertion of the shutdown rods. The heat absorption capacity of the RV is also included. Both cases, [

].

Figure 23. Time dependent relationship of fuel salt temperature and the RV heat exchange

[

]

Inspection and Testing

[

].

Maintenance

IRVACS is specifically designed such that minimal maintenance is required. If any component should fail, [ ].

However, any maintenance procedures will be determined during the detailed design phase.

Maintenance procedures shall be specifically developed, (during detailed engineering) to deal with the toxic nature of [ ].

Irradiated Fuel Auxiliary Cooling System (IFACS)

The Irradiated Fuel Auxiliary Cooling System (IFACS) [

].

Since this system is required to cool the irradiated fuel salt [

].

The primary safety-related purpose of IFACS is to remove decay heat from the irradiated fuel salt located in the FSST under variable conditions of the fuel salt amount held in the FSST, and the irradiated fuel salt decay heat loads, during accident scenarios when process cooling is not available.

Adequate heat must be removed to:

- [
-].

These represent the main functional requirements of IFACS, [ ].

The amount of irradiated fuel salt and decay heat to be removed from the FSST will vary over the life of the station, and is influenced by:

- [
-].

After 7 years of operation, the first batch of fuel salt [

]. When each Core-unit reaches its end-of-life, [

] within the FSST at end of plant lifetime.

IFACS will be capable of removing the maximum decay heat from the FSST, which is calculated to be [ ].

[

]. This is done through:

1. [
2.
3.
4.
5.
6.
] to radionuclide release through IFACS to the atmosphere, as IFACS acts as a Containment extension, since it penetrates the FSST and does not have isolation valves.
7. A draft chimney to promote draft airflow across the air-cooled heat exchanger.
8. A common inlet air duct].

There are [

], although it may be impaired and unable to continue operation.

As IFACS is a passive system, it does not require a power source. This also allows the IFACS loop to be always operating, and therefore its functionality can be continuously monitored and verified. [

].

Main interfacing systems with IFACS are the FSST, Containment, the FSST Vault and the RAB.

Secondary Coolant System (SCS)

The purpose of the SCS is to transfer the heat generated in the fuel salt to a non-radioactive salt called the Tertiary Salt.

The secondary coolant salt enters the Core-unit's vessel head to the PHX inlet headers and flows through the tubes of the PHX, removing heat from the fuel salt. From the outlet headers of the PHX, the secondary coolant salt exits out of the vessel head to the SCS pipes and then to the tubes of the Secondary Heat Exchangers (SHX), which transfer the heat load to the next loop called the Tertiary Coolant System (TCS). The secondary coolant salt is operated at a higher pressure than the fuel salt so in the event of a Primary Heat Exchanger tube leak, leakage is inward into the fuel salt rather than out of the radioactive fuel loop into the secondary cooling loop. The secondary coolant salt does not react with the fuel salt and does not cause any significant nuclear or pressure transient effects in the event of a leak.

When the secondary coolant salt is inside the Core-unit, it is irradiated by high gamma and neutron flux.

Traces of tritium as well as traces of neutron activated coolant isotopes could be in the coolant salt, and oxygen and moisture will be prevented from contaminating the mixture.

The operating fluid used in the SCS design is a molten fluoride salt. The salt selected is [ ]. The salt is a near-eutectic mixture with a melting point of [ ]. The components of this salt inherently do not produce significant amounts of tritium (i.e., much lower than in conventional nuclear power plants (NPPs); however, a tritium removal system will be installed, if necessary, to further reduce any tritium emissions to as low as reasonably achievable (ALARA).

As an overview, the following system design considerations provide a background for its generic features and physical configuration.

1. The SCS is designed to operate well below the nil ductility temperature of the major equipment and components.
2. When the coolant salt is inside the Core-unit, it is irradiated by high gamma and neutron flux. Traces of tritium, as well as traces of neutron activated coolant isotopes, could be in the secondary coolant salt. Oxygen and moisture will be prevented from contaminating the salt mixture. For these reasons, the SCS circulating loops are designed with high leak-tightness, high temperature resistance, ability to accommodate thermal expansion, and instrumentation and monitoring provisions to detect leaks and corrosion.
3. Typical piping in-line devices such as flanges, quick-connect couplings, and expansion joints are not permitted in this system since they have an operating experience history of leaking. Piping joints are all welded.
4. [
5.].

The SCS is comprised of [ ] and then to the Secondary Heat Exchangers, which transfer the heat to the TCS.

The SCS's [ ].

Figure 24 shows the SCS operating hot leg and cold temperatures. Hot and cold leg temperatures of the other heat transfer systems forming part of the overall heat transfer path are also indicated for context.

SCS temperatures are inputs to the SCS design.

Figure 24. Heat Transfer Loop Operating Temperatures

[

]

[

] may be pursued during the detail design phase.

Figure 25 displays a simplified flow diagram for one of the Secondary Coolant loops. The [

]. The Secondary Coolant Salt is on the tube of the SHX, and the Tertiary Coolant Salt is on the shell side. The SHXs are placed outside the Reactor Vault. The SHXs form the interface between the SCS and TCS. The SHXs are a permanent major component of the SCS and are not expected to be replaced with each Core-swap. As a consequence, [

].

Figure 25. Secondary Coolant System Simplified Flow Diagram (1 Loop of 6)

[

]

Tertiary Coolant System (TCS)

The TCS transfers the heat from the SCS through the SHXs to the Steam Generation System (SGS). The Tertiary Salt is comprised of [ ]. The Tertiary Salt leaves the SHXs area of the RAB and enters the Steam Generation Building to transfer the heat energy to the steam generation systems or process heat application.

Nitrate salts are excellent coolants, and the addition of the TCS adds relatively small and modest equipment, pumping power and related costs. The TCS is provided for the following reasons:

- The loop adds an additional barrier and inherent heat sink between the highly radioactive fuel salt and the non-nuclear Steam Plant and Turbine System.
- The loop acts as an additional barrier for tritium from entering the Steam Plant and Turbine System. A small amount of tritium is formed [ ] by nuclear reactions in the fuel salt.
- The TCS allows integration with the steam side and process side. The Tertiary Salt has a lower melting point [ ] than the feedwater temperatures [ ], to avoid salt freezing inside the steam generation systems during operational maneuvers and transients.

- In the event of any tube leakage from the Steam Generation System, the nitrate salts do not chemically react with steam, unlike fluoride salts, which would generate small but toxic amounts of hydrogen fluoride.

The nitrate tertiary salt heats a pressurized water loop by the use of steam generation systems, boiling the water to steam under pressure. Additional superheating from the hot tertiary salt provides dry, high quality, superheated steam of up to [ ] (due to the flexibility in the equipment, temperature and pressure can be varied as needed).

The TCS is insulated and contains isolation valves on the inlet and outlet pipes for operational and maintenance purposes. The TCS is designed so that it can be inspected and maintained as required.

[ ].

Off Gas Management System

The Off-gas management system is now included as a subset of the Irradiated Fuel System. See the discussion below.

Makeup Fuel System (MFS)

The main function of the MFS is to periodically add fuel salt (with a higher reactivity than the circulating fuel salt) to the circulating fuel salt to maintain the reactivity of the Core-unit and maintain the operating temperature of the Core-unit. The rate of makeup fuel salt additions will vary over the life of each Core-unit, with an average of [ ] of Makeup Fuel Salt. This equates to an addition rate of approximately [ ] per week, with a daily addition rate of [ ] on average. In addition to the amount of Makeup Fuel Salt added on an ongoing basis, each Core-unit after the second Core-unit will require an initial Makeup Fuel Salt addition to compensate for decay in the FSST.

The MFS is comprised of the following sub-systems and components:

1. Makeup Fuel Storage System:
    a. [
    b.
    c.
    d.].
2. Makeup Fuel Transfer System:
    a. [
    b.
    c.
    d.].

The Makeup Fuel Salt Canister and Fuel Salt Transportation Vehicle are not considered part of the MFS.

The Makeup Fuel Salt will be transported within the Makeup Fuel Salt [

]. The Makeup Fuel Salt will then be transferred [ ]. Before transferring directly into the Core-unit fuel salt load, the [ ]. An overview of the MFS, from the Fueling Room to the Core-unit, is shown in Figure 26. A flow diagram of the MFS can be found in Figure 27 (MFS Flow Diagram).

The makeup fuel salt is manufactured [ ], with a uranium enrichment of about 4.95%.

The MFS pipes are made of [ ] with an allowable thickness to accommodate corrosion over the plant lifetime. The materials selected for the isolation valves and pipe couplings, and compressors on the makeup fuel salt addition lines will be determined during Detailed Engineering in conjunction with a commercial supplier to ensure compatibility with the makeup fuel salt and gases produced by the Core-unit.

The MFS is designed to include provisions to periodically inspect and perform a walk- down of components (i.e., visual inspection) such as the attachment mechanism and isolation valves. [ ].

Figure 26. Simplified Diagram of the Makeup Fuel System

[

]

Figure 27. MFS Flow Diagram

[

]

All valves, conveyors, and couplings utilized for Makeup Fuel Salt transfer are [
]. Sufficient working space will be provided in the [

].

A preventative maintenance program will be identified during Detailed Engineering to determine the frequency and tasks required to maintain these MFS components.

Only [ ]. However, if
[
].

Additional detail about the system design and operation can be found in IMSR400-22500-DD-002, "Design Description Makeup Fuel System" (R-15).

Irradiated Fuel System (IrFS)

The main purpose of the Irradiated Fuel System (IrFS) is to transfer irradiated fuel salt between the various Core-units and Fuel Salt Storage Tanks (FSSTs), as well as to store all the irradiated fuel salt produced over the lifetime of the IMSR® within the FSSTs. At the [ ], the IrFS has provisions for [ ].

Due to the reliance on the [ ].

Design requirements for the IrFS are provided in IMSR400-22500-DR-002, Irradiated Fuel System (R-28) and IMSR400-22500-DR-005, Gas Management System (R-29).

The IrFS has two main functions:

1. Irradiated Fuel Salt transfer

A key function of the IrFS is to transfer fuel salt both to and from the Core-units and FSSTs after every seven- year Core-unit operational cycle, since irradiated fuel salt from spent Core-units is utilized as the starting fuel salt load for future Core-units.

The irradiated fuel salt transfer is facilitated through a [ ].

This also functions as a secondary method of providing a [ ].

2. Irradiated Fuel Salt and Off-gas storage

There is a need to contain and store all irradiated fuel salt and off-gases produced over the lifetime of the IMSR®, [ ].

The FSSTs are sized such that all spent fuel salt over the lifetime of the IMSR® can be contained within them. This includes the initial fuel salt load, as well as all makeup fuel salt additions over the plant lifetime.

As described in IMSR400-22500-ASD-002, Fuel Salt Storage Tank Design (R-72), makeup fuel salt is periodically added to the Core-unit over its lifetime to account for burnup. This results in a larger volume of fuel salt in the Core-unit at the end-of-life when compared to the start-of-life. Therefore, [ ].

The off-gases produced over the plant's lifetime are also contained and stored by the IrFS. This is done through the use of gas holding tanks, which store nearly all off-gases during Core-unit operation, [ ]. The gaseous sections of the Core-unit and FSST [

].

The IrFS consists of these main components:

1. Fuel Salt Storage Tanks
2. Gas Holding Tanks
3. Fuel Salt transfer lines and related components (valves, couplings)
4. Gas transfer lines and related components (valves, couplings, compressors)

The fuel salt transfer between a Core-unit and a FSST is facilitated by utilizing the [

].

Fuel salt transfer occurs [

].

All valves and compressors utilized in [

].

At the beginning of Year 1, initial fuel salt is transferred from the FSST to the new Core-unit, but no transfer from a spent Core-unit into the FSST is required. In year 56, irradiated fuel salt is transferred from a spent Core-unit into the FSST, but there is no new Core-unit to transfer a starting load of irradiated fuel salt into.

Gas Holding Tanks

Two Gas Holding Tanks are [

]. Each Gas Holding Tank has a volume of [ ], a design pressure of at least [ ] MPa, and dimensions of roughly [ ].

The Gas Holding Tanks will be connected by [

]. The transfer lines allow for [

], while also allowing all [

]. This allows for [

].

Each Gas Holding Tank will be [

].

The Gas Holding Tanks will [

].

The Gas Holding Tanks will also [

]. This will result in [

].

The Gas Holding Tank [

].

Between each Core-unit operation, roughly [

], will be used for removal of the excess gases. These pressure tanks have a diameter of [

]. These pressure tanks will be stored outside of the Reactor Auxiliary Building, but on the I-NPP site.

Fuel Salt Storage Tanks

Two FSSTs are provided as part of the Irradiated Fuel System. One tank is [

], which is designed to have an operational lifetime equal to the lifetime of the IMSR400, is determined to have a reduced lifetime.

The [

] is necessary.

Each FSST has the capacity to contain all fuel salt produced over the lifetime of the IMSR400, which will be [ ] salt. Since an empty Reactor Vessel (RV) has a volume of [ ], the design of the FSST will use the [ ]. Like the Core-unit, the FSSTs will be [

], which will be defined during design of the Containment System. As shown in Figure 28, each FSST, and its respective Containment, will be contained within its own Vault.

The irradiated fuel salt within the FSST will be held at or above [

]. During fuel salt transfer, the FSST will be [

] by the Irradiated Fuel Cooling System (IFCS).

Due to the radioactive heat load within the FSST immediately after transfer, the IFCS [

]. However, by one year into a Core-unit's lifetime, the IFCS will [

].

Safeguards for the FSSTs are ensured through multiple layers of defense. The FSSTs are within their own Containment volume, which is within its own Vault, which is within the RAB. The [ ].

Irradiated Fuel Salt and Gas Transfer System

Irradiated fuel salt transfer is performed as part of the Irradiated Fuel System and will be facilitated using fuel salt transfer lines, which will consist of [ ] piping. A representation of the configuration of the transfer lines, is shown in Figure 28 below. There will be [ ].

The fuel salt transfer lines [ ].

The fuel salt transfer line enters [ ] and molten fuel salt.

The gas transfer lines [ ]. After the fuel salt transfer is complete, [ ].

The gas transfer lines will be [ ]. The purpose of this is to [ ].

Figure 28. Irradiated Fuel Salt and Gas Transfer Configuration

[

]

The FSST will [

], radiolysis would occur in the irradiated fuel salt, whereby ionizing radiation splits the molecular bonds in the fuel salt, resulting in volatile and corrosive gases being produced, such as UF_6 and Fluorine gas (F_2). Differential thermal expansion of the FSST when compared to solidified fuel salt must not affect the structural integrity of the FSST.

As the FSST and solidified fuel salt cool, the FSST will thermally contract more than the fuel salt, as [ ] 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.

2. Localized freezing due to heat removal from the irradiated fuel salt in the FSST cannot affect heat transfer such that the localized temperature on the FSST wall will be outside of allowable temperature limits.

However, to avoid any potential of radiolysis, the idle temperature of the irradiated fuel salt in the FSST will be 200 °C. A heating system is provided to ensure the irradiated fuel salt stays at or above its idle temperature].

The composition of the irradiated fuel salt will be measured [

]. Other necessary measurements [ ].
The techniques for measurement [

].

Inspection and sampling of the off-gas will also occur [ ].

The fuel salt and off-gas transfer lines, FSSTs and Gas Holding Tanks are made of [ ] pipes with an allowable thickness to accommodate corrosion over the plant lifetime. The materials selected for the isolation valves, pipe couplings, and compressors on the fuel salt and gas transfer lines will be determined during Detailed Engineering in conjunction with a commercial supplier to ensure compatibility with the fuel salt and gases produced by the Core-unit.

The IrFS is designed to [ ]. This will occur within the [ ]. If necessary, [ ].

As the IrFS [ ].

Generally, there are [ ].

The isolation valves will be tested [ ]. Compressors will be tested by [ ].

The IrFS is designed to include provisions to periodically maintain components such as the valves, compressors and couplings associated with the fuel salt and off-gas lines.

All valves, compressors, and couplings utilized for irradiated fuel salt or off-gas transfer will be [ ].

The IrFS will be seismically qualified, and it must retain its structural and pressure boundary integrity during and following a Design Basis Earthquake, and therefore will be seismic Category A.

FSST Active Cooling System

The FSST will, at times, [

]. The FSST Active Cooling System (FACS) is designed to remove this heat to:

1. maintain irradiated fuel salt under saturation temperature to prevent potential boiling of the irradiated fuel salt contained in the FSST; and
2. maintain FSST and FSST components within their design temperature limits, thus maintaining the FSST pressure boundary.

The FACS, [

]. This is the main functional requirement of FACS,
[ ].

FACS requires a [

].

The amount of irradiated fuel salt and decay heat to be removed from the FSST will vary over the life of the station, and is influenced by:

1. [

2.

].

After 7 years of operation, the first batch of fuel salt [ ] will be transferred to an FSST from end-of-life Core-unit 1.

Fuel salt transfers from an end-of-life Core-unit to the FSST [
]. When each Core-unit reaches its end-of-life, [

] within the FSST at end of plant lifetime.

FACS is [

] within its design temperature limits.

As the heat load in the FSST will [

].

Otherwise, when a FSST is fueled with irradiated fuel salt, [

].

Operator action is [

1.

2.

].

FACS main interfacing systems are the FSST, Containment, the FSST Vault, and the RAB.

Gas Holding Tank Cooling System (GHTCS)

The Gas Holding Tanks will, at times, contain nearly all the off-gases produced by a recently defueled Core-unit. Gas Holding Tank integrity must be maintained to contain these off-gases, which are radioactively decaying and producing heat. The GHTCS, which is going through the design process, will be designed to manage this heat to ensure that the integrity of the Gas Holding Tanks is maintained.

Gas Holding Tank integrity is provided during all normal operation and AOO scenarios when heat load is high enough to require cooling, and therefore [

]. This is the main functional requirement of the GHTCS, in addition to maintaining a containment boundary.

For conservatism, it is assumed the GHTCS is [

] heat load.

As the heat load in the GHTCS will vary significantly over the plant lifetime, there may be [

]. Otherwise, the GHTCS will be [

].

[

]. Options for plant parameters used to regulate heat removal rate include, but are not limited to, [ ].

There are two [

1.
2.

].

The GHTCS main interfacing systems are the FSST Containment, the FSST Vault, and the RAB.

Hot Cell Cooling System (HCCS)

The primary purpose of the HCCS, which is going through the design process, is to maintain Hot Cell boundary integrity by removing decay heat from the Hot Cell upon [ ]. This is the main functional requirement of the HCCS, in addition to maintaining a Containment boundary.

[

]. Therefore, HCCS components which [

].

The design of the HCCS is performed in accordance with appropriate quality assurance program requirements. The HCCS [ ].

During normal operations, the HCCS will be in [

].

[

].

The HCCS will use an [

].

The HCCS interfacing systems are the FSST Containment, the FSST Vault, and the RAB.

FSST Vault Cooling System (FVCS)

The Fuel Salt Storage Tank (FSST) Support Structure (FSS) and FSST Vault will absorb heat mainly via heat

transfer from the hot FSST [ ], which is required for the following reasons:

1. [
2.
3. [ ] scenarios].

The failure of the FVCS [ ].

The FVCS is required whenever [ ] is not available.

[ ]. Options for plant parameters used to regulate heat removal rate include, but are not limited to, FSST Vault and FSS temperature.

The FVCS main interfacing systems are the FSST Vault, the FSS and the RAB.

Process Reactor Auxiliary Cooling System (PRACS)

The Process Reactor Auxiliary Cooling System (PRACS) is an active, process cooling system within the IMSR400. It functions as [ ] considered to be a part of the PRACS.

PRACS is [ ] is unavailable. This is the main functional requirement of PRACS.

PRACS cooling functions are [ ].

[

], as this will increase PRACS heat removal capabilities.

Preliminary analysis suggests that PRACS can [ ].

During at power operation, [ ]. During maintenance outages, [ ]. In this mode, the [ ].

[ ]. Options for plant parameters used [

[ ].

Further optimization is required but a preliminary layout sketch is provided below in Figure 29.

Figure 29. PRACS layout – side view looking North

[

]

V. 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® design. 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, channelization and separation rules are employed for the redundant systems and instrumentation to maximize independence. The design of the I&C architecture ensures a single failure of any component does not prevent any control, safety, or monitoring function from being performed. The general approach to channelization for I&C systems (and process actuator equipment) is to use two redundant channels when high reliability is required. There are [ ].

Description of Plant Control and Monitoring System

The Plant Control and Monitoring System (PCMS) of the IMSR400 nuclear power plant provides a common platform to implement fundamental safety related functions and control functions of the power plant operations. The PCMS comprises the equipment needed to control and monitor the main power production systems, auxiliary systems, and safety related systems. It obtains its inputs from instrumentation of the various interfacing systems and provides control signals to the device logic of the actuator components of the interfacing systems.

The PCMS provides the central control of the plant as directed by the operator through the operator interfaces to the PCMS. The layout, ergonomic design, and functional behavior of the operator interfaces stem from control facilities design requirements and the human factors program. The MCR is the center for all plant operations. From here, the operator can perform all control, monitoring and

safety related functions of the plant. In case of unavailability of the MCR, operators would move to the SCA to monitor and ensure that the plant remains in a safe state.

Figure 30 shows the Control Building and Reactor Auxiliary Buildings which house the control centers: a common MCR and two SCAs. The MCR is the center for all plant operations where the operators normally perform all control, monitoring and safety functions of the plant. Should the MCR become unavailable, the operators use the SCAs to monitor and ensure that the plant remains in a safe state.

Information on many of the manufacturer's instrumentation is proprietary, but OPEX information has been provided where available. Much of the Siemens technology is available in the public domain.

Figure 30. Control Centre Locations

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System Description

The PCMS is divided into the following sub-systems:

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The NPCS and the SPCS provide the [

] of the PCMS is shown in Figures 31 and 32. These figures present a complete configuration as an overview of the whole PCMS layout.

The use of the [

]. All key components involved in the IMSR400 production process [

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The fault-tolerant configuration is to be implemented with three key PCMS operating levels:

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Figure 31. Fault-Tolerant Configuration for PCMS Non-Seismically Qualified Portion

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Figure 32. Fault-Tolerant Configuration for PCMS Seismically Qualified Portion

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Operator Control-Monitoring Level

In the Operator Control-Monitoring Level, several operator consoles, termed in communication as data-access clients, can access data on one server. In addition, one client can access data on several servers through a pair of fault-tolerant terminal buses. It should be noted that the number and exact location of the consoles within the control rooms will be determined at a later stage in the design. What is shown in this example is typical of a process industry application of similar size and complexity. As shown in Figure 31, the operator consoles are installed in the MCR and the SCA. The SCA provides a backup for emergency operations should the MCR become unavailable for any reason. The operator consoles in the SCA are always in the operation-ready state regardless of the operating status of the MCR.

The control and monitoring operations of the IMSR400 plant are programmed to be carried out automatically, under the supervision of the plant operators by means of a number of Operator Consoles in the MCR as well as a number of Operator Consoles in the SCA for emergency monitoring. There is also an Engineering Console with which the plant designer has access to the configuration of the whole IMSR400 plant and can, under tight constraints, install updates to software and configurations as needed. There is also a Batch Console with which the plant operators can carry out batch operations for commissioning or testing. The system also includes a Maintenance Console with which the maintenance personnel can conduct the activities for routine maintenance or preventive maintenance.

The MCR is seismically qualified, but not the PDS/NPCS/SPCS partitions of the PCMS. For this reason, an additional SMS console is located in the MCR. This avoids the need to leave the MCR following a seismic event as critical parameter monitoring can continue with the seismically qualified console. The operator always has the option to go to the SCA and monitor critical parameters if necessary.

The number of consoles will be confirmed by Human Factors analyses but the architecture is very flexible in adding or removing consoles.

Automation Processing Level

In the Automation Processing Level, there are three control systems for processing the IMSR400 plant automation. They are the NPCS, SPCS and Secondary Control System as shown in Figure 32 above. The NPCS and SPCS have a similar engineering configuration which basically consists of redundantly configured automation systems for production critical operations.

The Secondary Control System consists of a group of redundantly configured automation systems for safety operations. Due to its safety significance, the Secondary Control System is independent of the NPCS and SPCS. However, its architecture may need modification such that communications between these independent partitions is not allowed (which would also prevent the propagation of failures between the partitions).

The automation systems execute all automated control and monitoring operations in the IMSR400 plant, under the supervision of the plant operators. The automation system is configured through the Engineering Console and can communicate with the operator consoles and the field devices. The automation system executes the data acquisition on the field devices and transports the data to the servers for display on the operator consoles under the management of the operators.

Field-Devices Interfacing Level

In the Field-Devices Interfacing Level, the use of field bus technology enables the use of "smart" transmitters that have built-in self-monitoring features for early detection of sensor failure.

The field devices are to be used for the IMSR400 operations, either with simple control/monitoring

mechanisms or with advanced signal-processing capability and communication capability. Depending on the communication types of the devices, they can be connected to the automation systems using I/O modules, as shown in Figure 32 above.

Greater details about the design can be found in IMSR400-24500-DD-01, Design Description Plant Control and Monitoring System (R-22).

Reactor Trip

There is no automated Reactor Trip system in the IMSR400, and therefore no associated instrumentation or control instrumentation is required.

All other instrumentation systems required for safety:

Pump Vibration monitoring

Pump vibration monitoring will be performed for all large pumps and compressors in the IMSR400 system to provide preemptive warning of equipment failure or required maintenance.

Leak Detection

Leak detection and integrity monitoring are crucial to preventing failures within the IMSR400 system, whether they be hazardous to personnel or economic losses. Monitoring of the RV outer surface and the area between the RV and GV is necessary to detect the integrity of the vessel wall itself, and to ensure that no leaks have occurred from the RV into the GV, and from the GV into the Reactor Vault. Inside the RV, it is necessary to monitor the Thimble integrity and to be able to detect if the mechanical seals on the IPS begin to leak. Both leaks would result in unwanted ingress of molten, radioactive salt. It is also desired to monitor the graphite moderator surrounding the Core to ensure that the barrier between the fuel salt and the boron carbide absorber layer has not been breached.

The containment boundary is monitored to ensure that it is at a negative pressure relative to the rest of the RAB (to mitigate the spread of airborne radioactive material) and to ensure that there is no significant oxygen ingress. These factors will be covered by air pressure monitoring and oxygen concentration monitoring, respectively.

The IRVACS and SCS loops are monitored to ensure that there is no ingress of fuel salt through the EHX1 or PHX, respectively. The IRVACS, SCS and TCS loops are all monitored to ensure that their respective coolant salts do not breach the vessels and pipes that make up their respective systems. The off-gas and fuel transfer lines between the RV, FSST and Hot Cell are monitored for leaks as well, to ensure proper containment of irradiated or fissile material.

During the Core Swap procedure, a close eye is kept on the conditions inside of the contamination boundary to ensure that there is no significant leakage of gases out of the boundary, and that there is no excessive contamination of Containment by the Core Swap process.

Criticality Accident Alarm System (CAAS)

The CAAS exists to monitor activity of nuclear materials stored outside of Containment, where personnel are allowed access. In the case of an out-of-core criticality (OCC) event, this system triggers an audible and visual local alarm in the affected area and alerts the PCMS that an event has occurred. This system is intended as defense-in-depth against OCCs, which are categorized as BDBAs due to the other methods employed by the IMSR400 to prevent them from occurring, such as physical separation and low enrichment of fuel. Failure of this system to notify personnel in the affected area cannot be tolerated as

it could lead to loss of life.

The CAAS will feature a central panel, which monitors radiation levels in any potentially affected areas via 3 or 4 local neutron detectors, situated to cover the entirety of their area. These 3 or 4 detectors operate on a voting system to mitigate the risk of false alarms, and to provide redundancy against single detector failure. When an OCC event is detected by the sensors, the central panel will trigger the alarm automatically, with the specific affected area sent to the operator in the MCR. The detectors will be organic scintillators, which are robust, have a good discrimination between neutron and gamma radiation, and have a short decay time which allows them to count effectively in high radiation environments such as those created by an OCC event. This ability to continue to monitor elevated radiation levels is especially important for Post-Accident Monitoring (PAM). Currently, there will be three monitored areas in the RAB:

1. Fuel transport route
2. Fuel Salt Storage Room.
3. New & Makeup Fuel Room.

The neutron detector range is required to determine the exact number of areas, which will be determined upon device selection in the Detailed Design phase. As the RAB layout evolves, the number and location of the areas may also change. As it stands, using the terminology from Section 5.3.3 of IMSR400-30805-REP-018 (R-71), the CAAS will monitor Areas 4 and 5 at EL 100M, as well as the corridor and loading bay leading up to those areas. Figure 33 shows the currently identified Areas of Interest (AOIs) in the RAB.

The CAAS will be seismically qualified, and thus connected to a seismically qualified uninterruptible power supply. It will also provide constant feedback of reactivity and radiation levels to the PCMS, along with any self-diagnostics that would indicate errors, component failure, and power loss. The communication between the CAAS and PCMS is intended to be one-way, in that the PCMS will not be able to send commands to the CAAS panel; as such, cybersecurity concerns around remote access to the CAAS panel are addressed by the PCMS design. Software validation of the CAAS itself is expected to be completed by the system supplier.

There is no intention to procure a redundant system to the CAAS, due to the low compounded probability of an OCC event occurring alongside an outage of the primary CAAS. Regular (monthly) testing is intended to ensure that the CAAS remains in working order throughout its intended lifetime.

There is the potential for an interlock between the CAAS and the ventilation system in the affected area. A triggered alarm would close ventilation to the area to prevent the spread of any airborne radioactive material. Minimal airborne material is expected to be generated in an OCC event. Once triggered, the local alarm cannot be reset remotely, but must be reset manually at the central panel. The system is expected to be always online.

Figure 33. Currently identified AOIs in the RAB

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Environmental Qualification Envelope

The process instrumentation is present in many different areas of the plant, leading to a broad range of conditions depending on the function of the sensor. Environmental conditions tend to become less harsh as distance from the Reactor Vessel (RV) increases.

The most severe conditions are encountered within the RV and bounded by the Guard Vessel (GV). Temperatures there can exceed [ ]. Instrumentation must therefore be qualified for both high temperature operation, and potentially long-term operation in a high radiation field. Systems that fall under this category are the RV itself, the Shutdown Mechanism (SDM), Integral Pumping System (IPS), Primary Heat Exchanger (PHX), Fuel Salt Storage Tank (FSST), Hot Cell, Internal Reactor Vessel Auxiliary Cooling System (IRVACS) and the Graphite Moderator.

The RV and GV are located within the Reactor Vault. The temperature and radiation levels are high here as well, potentially reaching [

]. Operations within the Containment boundary are to be carried out by robotic or otherwise automated means, as the conditions would be unsafe for personnel even with protective equipment. Vault/Containment Cooling, Gas Management and Irradiated Fuel, Initial Fueling, Makeup Fueling, the IFACS, Spent Core-Units and the Core Swap system all operate within this space.

There are also systems outside of Containment which require process instrumentation and operate in much milder environments. The most significant change outside of Containment is a lower radiation level, with temperatures elevated but not to the extent of those within the Reactor Vault. Such systems include the SCS and TCS loops, which are located in separate rooms in the RAB, or the RAB HVAC, which is connected to the Reactor Vault via ductwork.

Table 1 outlines the environmental requirements (temperature and radiation) for the IMSR400 sensors. As the majority of the detailed calculations of temperature and radiation levels are pending, the following categorization system is defined for the severity of environmental conditions:

- Harsh: Temperature and/or radiation levels are extremely high in the area. Personnel access is prohibited.
- Moderate: Temperature and/or radiation levels are high in the area, but situational use of appropriate protective equipment can mitigate the effects. Personnel access is restricted.
- Mild: Temperature and/or radiation levels are nominal in the area. Personnel access is not restricted on a safety basis.

The final qualification envelope of a system is the highest of these environmental categorizations in which the system is required to perform its functions.

Table 1. Environmental Requirements for IMSR400 Process Instrumentation, based on (R-73)

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VI. Electrical Systems

The IMSR® Electrical Power System provides interconnection of the plant to the power transmission grid plus distribution of power to the many plant electrical loads. This description presumes that the IMSR® is designed as an electrical power plant although the Steam Plant could instead be a process application like ammonia, desalination, or hydrogen. The system is explained in detail in the Design Description, IMSR400-24000- DD-001, Electrical Power Systems (R-26). Figures 34 and 35 provide a schematic of the electrical system design.

The electrical power from the steam turbine generator is directed to the transmission grid and electrical power is imported to the power plant from the transmission grid to service various plant loads. The electrical power distribution to the generating plant consists of multiple classes of power (normal, standby, and uninterruptible), each with at least two divisions of electrical power distribution (odd and even) to meet the reliability and safety requirements of the plant. The uninterruptible power for the nuclear systems instrumentation is triplicated.

The on-site IMSR® Electrical Power System is divided into four sub-systems in accordance with the different requirements of the plant electrical loads:

Normal AC Power Supply

This sub-system provides electrical power to the assigned AC loads that will tolerate an interruption in the power supply. The main AC power is normally supplied from the unit turbine generator via a unit service transformer (UST) that has two separate secondary windings designated odd and even that supply power to their respective division of electrical power distribution. Upon the loss of power supply from the main generator, main AC power can also be supplied from offsite power via the switchyard and system service transformer (SST) which also has two secondary windings.

Standby Power Supply

The preferred source of AC power is from the grid or unit power; however, the AC power system also includes standby AC power sources. Protective relays detect loss of the normal AC power supply to the electrical power systems and automatically start an electrical generator for the standby electrical power supply of the affected division.

Uninterruptible Power Supply

This sub-system supplies DC and low-voltage AC loads from the AC power system via rectifiers/inverters, and without interruption from batteries when AC power is not available.

Alternate AC Power Supply

This sub-system consists of dedicated connection points for mobile and fixed diesel generators. Its primary function is to support the Uninterruptible Power Supply if the Standby Power Supply is not available.

Figure 34. Simplified Single Line Diagram - HV and MV

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Figure 35. Simplified Single Line Diagram – LV

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A more in-depth description of each sub-system within the IMSR® Electrical Power System is provided below:

Normal Power System

The Normal Power System is subdivided into two redundant power distribution divisions, designated ODD and EVEN. The two normal supply divisions are supplied from the turbine generator and the switchyard. The supply from the turbine generator is via the Unit Service Transformer (UST) and the supply from the grid is via the SST. This allows the facility loads to be supplied from either [

] loads.

This two-division configuration allows for a single fault in one division with the remaining unfaulted division available to supply power to the connected loads.

The major loads supplied by the Normal Power System are:

1. Boiler Feedwater Pumps
2. Condenser Cooling Water Pumps
3. Condensate Extraction Pumps
4. Coolant Salt Pumps
5. Solar Salt Pumps
6. Coolant Salt Heaters
7. Solar Salt Heaters

Standby Power System

The arrangement of the Standby Power System is similar to the Normal Power System in that there are two redundant divisions of power distribution for ODD and EVEN. Two [ ] standby generators are provided to meet the reliability requirements of the facility. The Standby Power System is designed to function in the presence of a single failure. The ODD standby power system is normally supplied from the ODD normal power system and the EVEN standby power system is normally supplied from the EVEN Normal Power System. In the event of a loss of normal power supply, the Standby Power System is isolated from the Normal Power System, and the standby generators are automatically started and connected to the Standby Power System distribution. The loads are re-energized according to the sequence logic in discrete steps which will minimize the magnitude of voltage and frequency transients impressed on the Standby Power System due to the starting of large motors. Where variable flow is required by the process demand, variable speed drives (VSDs) are utilized. The VSDs are designed to have low input current harmonics so that they are compatible with the standby generators.

The major loads supplied by the Standby Power System are:

1. Auxiliary Boiler Feedwater Pump
2. Service Water Pumps
3. Fire Water Pumps

4. Fuel Salt Pumps
5. Fuel Salt Heaters
6. Reactor Vessel Heaters
7. Air Compressors
8. Uninterruptible Power System

The 4.16 kV standby power buses have the capability to be interconnected such that one standby generator can supply all of the standby loads connected to both buses. The 4.16 kV standby power buses are interconnected [ ] each bus. This connection arrangement permits isolation of each bus and facilitates maintenance.

Part of the Standby Power System is seismically qualified. This portion contains ODD and EVEN divisions of power distribution. Each division is normally supplied from the Standby Power System via a manually operated transfer switch. The other source of supply to the transfer switch is from a permanently installed connection box which is provided for each division of power distribution. Each division is installed in separate rooms and is independent of the other with one exception. The divisions can be connected together via interlocked circuit breakers so that one alternate power supply can supply both ODD and EVEN divisions while isolated from the normal power supply. Interlocks are provided such that only one source of power can be connected to a bus at any given time.

Uninterruptible Power System

The uninterruptible power system is sub-divided into seismic and non-seismic qualified systems. The seismic and non-seismic qualified systems are further subdivided into [ ] systems.

The topology for a division of uninterruptible power system distribution is [ ]. The AC input supplies power to the [ ].

Each uninterruptible power distribution division contains a DC uninterruptible power division and an AC uninterruptible power division. The nominal voltage of a DC uninterruptible power division is 250 VDC.

A DC uninterruptible power division contains:

- two 100 % redundant rectifiers
- DC switchgear
- batteries
- a connection terminal for testing the batteries with a load bank.

An AC uninterruptible power division contains:

a. A DC to AC inverter
b. A static transfer switch
c. A manual bypass switch
d. A bypass voltage regulator
e. AC distribution panels

The DC switchgear supplies power to DC loads including the DC-AC inverter. The inverter supplies AC power to the Uninterruptible Power AC buses via [

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between the two sources. The manual bypass switch is make-before-break to enable a smooth transfer between the static switch and the manual bypass switch.

The non-seismic qualified uninterruptible power system in the nuclear plant supplies power to the following loads:

- NPCS/PDS
- Emergency Operation Lighting in the CB and RAB
- Emergency HVAC to cool the NPCS/PDS hardware
- Communications system
- Control power for switchgear located in the CB

The seismic qualified uninterruptible power system supplies power to the following loads:

1. Post-Accident Monitoring instrumentation
2. Secondary Control System/SMS
3. Emergency Operation Lighting in the MCR and SCA
4. Emergency HVAC to cool the Secondary Control System/SMS hardware
5. Emergency HVAC in the MCR/SCA
6. Emergency communications system

The non-seismic qualified uninterruptible power system in the steam plant supplies power to the following loads:

- SPCS
- Control power for switchgear located in the TAB.
- Control power for the switchyard

The batteries are flooded lead acid, and each battery is located in a ventilated battery room. Ventilation is provided to prevent the build-up of hydrogen gases during charging. A separate and independent battery room is provided for each division of battery. A separate and independent switchgear room is provided for the rectifiers, inverters and DC switchgear associated with a division of power distribution.

The non-seismically qualified Uninterruptible Power System has a battery mission time of [

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The DC Uninterruptible power divisions are ungrounded and incorporate ground fault monitoring. As the system is ungrounded, it can still operate with only one ground fault. For this reason, the reliability is considered higher than a grounded system. However, the ground fault should be located and fixed as soon as possible in case a second ground fault should develop.

Each DC uninterruptible power division contains a battery. Batteries require periodic testing to verify

their capacity. In order to facilitate the battery test, the battery is connected to the DC switchgear via a double pole double throw switch. With the switch in the normal position, the battery is connected to the DC switchgear. With the switch in the test position, the battery is isolated from the switchgear and connected to the battery test terminal. The test terminal is provided to allow connection of a load bank in order to perform a battery discharge capacity test.

Alternate Power Supply

The Alternate Power Supply consists of two seismically qualified diesel generators located in the Secondary Control Area which is also seismically qualified.

The Alternate Power Supply is normally not connected to the power distribution system. However, permanent connection boxes are provided. Portable power cables with color coded camlock connectors are used to connect the Alternate Power Supply generators to the provided connection boxes. Each connection box for the Alternate Power Supply has a means to confirm phase rotation. The connection boxes are located such that they are easily accessible for use with on-site power sources and off-site mobile power sources. The Alternate Power Supply generators are [ ] standby power distribution.

The Alternate Power Supply is designed to be tested at least once a month by means of a load bank. The load bank is connected via portable power cables with camlock connectors.

Black Start (Optional)

As a design option, it is possible to black start the facility with a dedicated onsite power generation source. This power generation source is black-start capable which means that it can start up without any external power supply or service. [ ] black-start generators are provided to meet the reliability requirements of the facility black-start function. The black-start generators are connected to the 4.16 kV normal power system buses, one for each bus. However, they do not supply the standby power system buses as there are standby generators for that purpose.

The start-up loads include the Boiler Feedwater Pumps (BFP) which are the largest motors in the facility and have an estimated rating of [ ]. The BFPs are connected to the normal power system via Variable Speed Drives (VSDs). The VSDs serve to reduce the motor starting current to minimize the voltage transients impressed upon the power distribution system whether supplied from the off-site power system or the black-start generators. In addition, the VSDs allow for variable feedwater flow and will eliminate a feedwater bypass valve for each pump. The variable speed drives are designed to have low input current harmonics, so they are compatible with the black-start generators. The estimated size of the black-start generators is [ ] and they are selected to be diesel.

VII. Core-unit Description

The purpose of this section of the report is to describe the proposed Integral Core-unit design chosen for the IMSR400. A key innovation of the IMSR400 is a sealed and replaceable assembly called the Integral Core-unit, or "Core-unit" which contains the graphite Moderator, circulating Fuel Salt, Primary Heat Exchangers (PHXs), Integral Pumping System (IPS) and other Structures, Systems or Components (SSCs) required to sustain and control the nuclear fission process. Each of these SSCs will be discussed in greater detail below. Figure 36 presents an overview of the Core-unit. The replaceable IMSR® 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.

Specifically, the Core-unit consist of the following major SSCs:

a. Reactor Vessel (RV)
    i. RV (Shell, Upper Head, Lower Head)
    ii. [ ]
b. Core-unit Internals
    i. Graphite Moderator
    ii. Reflector
    iii. Shielding
    iv. Internal Structural Steel
    v. Upper Hold-down Plate
    vi. Lower Support and Flow Guide
    vii. Chimney
    viii. Downcomer
c. Integral Pumping System (IPS)
d. Primary Heat Exchangers (PHX)
e. Core-unit Externals
    i. Guard Vessel (GV)
    ii. Heater
    iii. Insulation

Figure 36. IMSR400 Integral Core-unit Section View

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Additional components within the Core-unit include instrumentation, Internal Reactor Vessel Auxiliary Cooling System (IRVACS) components (such as EHX1), and the Fuel Salt and Off-gas transfer lines. The Core-unit has an outer diameter of [ ] and a height of [ ]. Piping connections associated with the Core-unit are provided for the:

[

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Reactor Physics and Reactivity Control

The IMSR® in-core components (nuclear fuel, moderator, coolant, and structural material) are arranged in a configuration to safely support a controlled, self-sustaining nuclear chain reaction to reliably generate the rated heat energy (442 MWth) per unit for seven years. In addition, criticality cannot occur anywhere in the plant other than inside the Core-unit (out- of-core criticality assessment). Physics assessments in support of these statements include:

1. Core neutron flux and power distribution.
2. Temperature feedback reactivity (due to fuel salt density changes, doppler effect, and moderator temperature).
3. Reactivity depth for reactor shutdown and Guaranteed Shutdown State (GSS).
4. Neutron fluence limits for in-core components (especially the moderator and in-core steel components).
5. Impact of voiding (e.g., bubbles) on core reactivity.
6. Fueling (initial fuel load and make-up fueling and the required uranium enrichments).
7. Fuel utilization.
8. Core optimization to define core configurations that are consistent with the design requirements and optimize the performance of the design.
9. Analysis of fission products and off-gas production over the life of the reactor and fuel.
10. Decay heat loads from the fuel, in-core and in storage.
11. Reactor response to power maneuvers (e.g., load following), transients and accidents.
12. Assessment of out-of-core criticality (OCC).

The qualification of the physics analyses is based on proven practices for validation and verification, using acceptable codes and standards, and operating experience from other operating reactors. The analytical methods used in the nuclear design (including those for predicting criticality, reactivity coefficients, burnup and stability) as well as the database and nuclear data libraries used for neutron cross-section data and other nuclear parameters (including delayed neutron and photoneutron data and other relevant data) have been validated against the benchmark experiments used in computational

reactor physics. Also, physical tests to further validate these codes are planned at test facilities.

There are three main ways to control reactivity when the IMSR® core is critical:

1. the negative temperature reactivity coefficient affected by core average temperature changes.
2. make-up fueling, and
3. activating the Shutdown Mechanism to shut down the reactor.

The IMSR® inherently controls reactor power to demanded power in the short term, without automatic manipulation of a reactivity control device. This feature is due to the highly responsive negative core temperature coefficient of reactivity. In the long term, reactivity is controlled by routine additions of small amounts of fuel salt. Secondly, there are no restrictive neutron flux limits in the molten fuel salt core compared to limits in traditional water-cooled reactors related to fuel sheath integrity. Hence, neutron flux can be allowed to transiently increase by a significant amount without any negative effects on integrity of the core. For these reasons, the IMSR® does not have any control rods nor automatic flux control algorithms for short term power control. The core temperature can be viewed as the control device which naturally adjusts itself so that reactor power matches demanded power in a relatively short period of time. The functions of the power control system are to enable the reactor to naturally match the power withdrawn by the Steam Plant to ensure salt temperatures stay within their integrity limits.

Over the lifetime of a Core-unit, burnup of fuel salt and fission product neutron absorption will result in a reactivity decrease. Due to the negative temperature reactivity coefficient of the Core-unit, this would require a decrease in fuel salt operating temperature to maintain fission power. To maintain the reactivity of the Core-unit, as well as maintain it within its operating temperature range, Makeup Fuel Salt with a higher enrichment than the circulating fuel salt load is added to the Core-unit(R-76). Figure 37 depicts the total fuel salt requirements over the plant life. These additions are made by the Makeup Fuel System (R-15, 77).

A partial fuel recycle strategy has been developed for the IMSR400 termed mixed [

Transfer between [

]. This is discussed

further in (R-76).

Table 2. Core-unit Physics Design Description

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A strong negative temperature reactivity coefficient is a fundamental feature of the IMSR400. The reactor core power follows the turbine load, by virtue of its strong negativetemperature reactivity coefficient and rapidly responds to load changes (R-76). This makes the system easy to control and eliminates the need for in-core reactivity control devices. Power changes are made simply by reducing or increasing feed water flow rate, in conjunction with a similar reduction or increase in the coolant salt flow rate (R-76).

Removing less heat from the Core-unit heats up the Fuel Salt and moderator creating negative reactivity which tends to lower reactor power to match the reduced demand for power. Conversely, removing more heat cools the Core-unit and creates positive reactivity that tends to raise power to the new demand level (R-76).

The strong negative reactivity coefficient is an inherent safety feature that ensures safe control of the reactor during Normal Operation (NO), and during any Anticipated Operational Occurrences (AOOs) or Design Basis Accidents (DBAs) that would tend to increase reactor power or overheat the Core-unit (R-76). In addition, to ensure shutdown of the Core-unit as required during NO, for maintenance, or to protect plant assets, shutdown rods are provided which are inserted either manually or automatically through the guide thimbles in the center of the core and provide sufficient negative reactivity to ensure shutdown for an extended period of time to establish a Guaranteed Shutdown State (GSS) (R-76). A second GSS can be established by defueling the Fuel Salt from the Core-unit to the FSST (R-76).

The neutron fluxes are an important indicator of the core power distribution during full power steady state operation (R-76). The reactor operation can be monitored by the neutron flux, and the neutron flux changes can be detected if an abnormal operating occurrence happens. The neutron fluxes are also important to calculate neutron fluence for reactor material analysis (R-76). Figure 38 depicts the fast neutron flux and thermal neutron flux distribution in the radial direction, at the center line of the core. Figure 39 depicts the same along the vertical center line.

Neutron fluxes within the Core-unit at other locations may be found in the Core-unit Physics Design Description (R-76) along with further details of the Core-unit Physics behavior.

Figure 37. Fuel Salt Required over the 56 Year Plant Life (8 Core-units) (R-76)

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Figure 38. Neutron Flux in fresh initial core along the center x axis (R-76)

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Figure 39. Thermal and Fast Flux for the initial core along the z vertical center axis (R-76)

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Table 3 provides a summary of the major core characteristics for the optimized core:

Table 3. Characteristics of Physics Optimized Core

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The three regions of the [

] of the reactor core. Figure 40 below shows the fine mesh flux and temperature distribution through the center line of the core.

Figure 40. Fine Mesh Flux and Temperature Distribution

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As discussed above, the IMSR® core has inherent negative reactivity coefficients of temperature throughout the life of the core. This is shown in Figure 41 below.

Figure 41. Inherent Negative Reactivity Coefficients (R-76)

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Out-of-core Criticality (OCC)

The reactor physics design of the fuel systems ensures that out-of-core criticality cannot occur, with a large safety margin, for any likely configuration during fuel handling for fuel makeup and storage, or fuel salt removal from core and storage, and as a result of DBA and BDBA accidents such as external leakage of fuel salt from the core.

Reactor Physics Assessment Document IMSR400-30500-ASD-020, Out-Of-Core Criticality Assessment for IMSR400 (R-61) describes the preliminary results of the OCC analyses. This assessment shows that out-of-core fuel remains deeply sub-critical for fuel handling (including accidental drop of both make-up fuel tanks), Make-up Fuel storage/transport, and on-site storage of spent fuel. Despite this, criticality alarms will be installed in areas where it is possible for criticality to occur. The criticality alarms are described in the design set of documents for the Criticality Accident Alarm System (R-40,41).

Reactor Thermal Hydraulics

The IMSR® reactor core contains moderator material (graphite), which forms vertical fuel salt channels within its volume. The fuel salt gets circulated through these vertical channels and the fissionable material within the fuel salt undergoes nuclear fission. The majority of the fission heat gets deposited at fission sites (meaning within the fuel salt). The remaining heat gets deposited into the moderator (graphite) material via fast neutrons, beta, gamma, etc. and some heat gets deposited outside the reactor core via delayed neutron precursor (DNP) and fission products (decay heat). The fuel salt within the graphite core acts as a fuel to generate the fission heat, and it also acts as a coolant to the in-core material, removing the deposited heat out of the graphite, thimble, shield, and reflector. In addition, the fuel salt acts as a coolant for the Core-unit internals that receive decay heat deposition.

The Core-unit thermal hydraulics design involves simulation and analysis to ensure safe and efficient removal of the prompt heat, delayed heat, and decayed heat from the entire Core-unit to maintain integrity of the Core-unit and its internals during all plant states. The objectives of the reactor core thermal hydraulics design are as follows:

- To assist the nuclear chain reaction to generate 442 MW_{th} by ensuring continuous fuel salt circulation through the graphite core.
- To ensure that the in-core components (fuel salt, graphite, thimbles, and shield) remain within their safe operating temperature during all the plant states.
- To ensure adequate natural circulation of the fuel salt in the core to remove the decay heat; and
- To minimize the fuel salt pressure drop through the core.

The 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. The analytical method and modelling techniques used in the thermal hydraulic design are based on the standard theories of fluid dynamics and heat transfer.

Figure 42 displays simplified flow paths for the thermal hydraulic flow simulations. The analysis and simulations are performed independently in two different codes, [

]. Using the results of these two codes, code-to-code validation is performed to acquire higher confidence on the simulation results.

For the Core-unit thermal hydraulics design, the Core-unit height is divided into [

] overall Core-unit thermal hydraulics design.

The reactor graphite core is supported by [
]. The graphite core channels have [

]. This would result in the [
]. Due to this phenomenon, the [

] temperatures.

Table 4 provides a summary of the key results from the current Thermal Hydraulic [
] at steady state (R-79). These results are used as inputs for the design of individual SSCs and inform where optimizations of the design should beinitiated in ongoing work. Modelling will continue to provide updated figures as the design evolves during Detailed Engineering.

Table 4. Steady State Thermal Hydraulic Modelling Major Outputs (R-79)

[

]

Figure 42. Core-unit Thermal Hydra Flow Path and Main Components Hydraulic Model (R-79)

[

]

Reactor Vessel

The RV is a cylindrical pressure vessel with ellipsoidal heads which houses the Graphite Moderator, PHXs, Emergency Heat Exchanger 1 (EHX1), circulating Fuel Salt, IPS, and supporting SSCs. Along with the PHX tubing, the RV forms the primary, nuclear boundary during NOs, AOOs, and DBAs. With no penetrations below the maximum Fuel Salt liquid level, primary circulating line break type accidents are eliminated. In addition, the vessel operates at low pressure, is conservatively designed, and made of high-quality alloys of construction.

Safety analysis will demonstrate RV failure does not occur in any DBA. However, containment surrounds the RV in the event of a Beyond Design Basis failure of the vessel, to catch and contain any leaked fuel

salt.

The RV operates in the following conditions:

- Molten Fluoride Fuel Salt,
- Low pressure (With short term increases during transients or fuel transfers),
- High temperature, and
- Significant neutron and gamma flux.

Because the RV is completely sealed during the 7-year operational life of the Core-unit it must operate reliably and without maintenance. A limited nominal design life of 7 years before a Core-swap is intended to mitigate the effects of these challenging environmental and operating conditions. Due to its simple shape and thickness, the RV is readily constructible using conventional metalworking techniques.

The schematics of RV head and shell with its general dimension are provided in Figure 43 and Figure 44. Table 5 summarizes the most important parameters describing the RV. The RV will be monitored and regularly inspected through [ ] solutions. RV preliminary temperature limits have been established based on conservative safety margins for thermal creep failure of [ ]. The RV also sees significant neutron flux. The flux must be below the alloy embrittlement limit, and the limit depends on the alloy chosen. In addition, cobalt and other high activation sources must be kept as low as possible in the alloy.

These limits are currently approximate and must be confirmed to be appropriate as part of the R&D and engineering phases. Research and development activities are in progress investigating the RV materials of construction. Items incorporated in these research activities include the effect of irradiation on the material properties and the determination of the corrosion and erosion effects on the material due to the working fluid of the RV.

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

- Temperature
- Pressure
- Neutron flux
- Fuel Salt level

Table 5 summarizes the most important parameters describing the RV.

Table 5. Reactor Vessel Design Description (R-81)

Parameter	Value	Basis
Primary loop architecture	Integral	Integrating all primary components into the RV contains the fuel salt within the RV and eliminates circulating line breaks.
Vessel shell	Cylindrical	High strength, ease of fabrication
Vessel bottom head	Ellipsoidal	Higher strength than flat bottom, needed to hold weight of salt, lower fuel salt volume than hemispherical bottom.

Parameter	Value	Basis
Vessel top head	Ellipsoidal	Higher strength than flat head.
Normal Operating Temperatures	[ ]	Salt inlet/outlet temperatures with adequate margin above the fuel salt freezing point.
Normal Operating Pressure[7]	[ ]	Vessel normally operates well below this pressure, however; some operations or transients can increase internal pressure. A full evaluation is ongoing.
Height	[ ]	Fits all components plus fuel salt (Including Makeup Fuel additions and thermal expansion) and sufficient cover gas/off-gas delay volume with margin. Provides sufficientnatural circulation driving head upon loss of forced circulation.
Outside Diameter	[ ]	Initial RV dimensions were based on a parametric study to optimize the Core-unit physics and thermal hydraulic behavior.
Material	[ ]	Other materials are still under consideration, but due to the material data available for [ ] it is currently considered the best option. Therefore, the preliminary analysis isbased upon the RV shell and heads being constructed of [ ]. The material properties used in the analysis of the RVare taken from the appropriate tables of ASME Boiler and Pressure Vessel Code Section III, Division 5 and Section II Part D (R-80)

This RV boundary:

- Contains the Fuel Salt and some high-activity off-gas.
- Provides a flow circulation path for the fuel salt; and
- Provides a support (anchor point) for the core internals.

Figure 43. RV Head Detail (R-19)

[

]

Figure 44. RV Side View (R-19)

[

]

Thimbles

Thimbles are [

] (R-82).

Operating experience from MSRE was considered in the [

]. Additionally, [

].

Figure 45. Preliminary Design of Thick-Walled Thimbles

[

]

Liquid Fuel Salt

The IMSR® is a liquid-fueled reactor, which operates by fissions of low-enriched uranium tetrafluoride (UF_4), dissolved in a molten primary coolant comprised of a fluoride salt-mixture. The major purpose of the fuel salt mixture is delivery of the low-enriched fissile uranium into the core for heat generation through the sustained chain reaction and subsequent transportation of the heated salt to the PHXs. The concentration of uranium and reduction potential of the salt system is a subject of continuous monitoring and adjustment in a timely manner over the 7-year operational lifetime. The lower liquidous 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. The fluoride fuel salt mixtures offer high potential for nuclear applications as they generally have the following essential characteristics:

[

]

Greater details of the Fuel Salt properties are described in IMSR400-08410-MDS-001, Fuel Salt (R-71).

IMSR® fuel salt is comprised of [ ]. The fuel salt is a green solid at room temperature and dark green liquid at temperatures above [ ] °C. As other experimentally studied comparable fluoride salts, the fuel salt is expected to behave as a single-phase Newtonian fluid (excluding fission gas bubbles) and be radiolytically [ ] stable.

The fuel salt data was estimated based on the experimental data reported by ORNL for the analogous salt mixture, [ ]. This estimated data (Table 6) is presently used in the design and is subject of further confirmation by measurements under the TEI quality standards for confirmatory testing. The justification of using this preliminary data in the design is supported by the recent studies conducted by Argonne National Laboratory revealing that the thermo-physical properties were comparable across several [ ] mixtures with varying concentrations.

Table 6. Estimated Properties for IMSR400 Fuel Salt and Experimentally Derived Correlations for [ ]

[

]

Fuel salt [

].

At elevated temperatures, interaction of [

].

The [ ], whereas below [

].

The chemical specifications of the initial and make-up fuel salts and additional quantity limits for selected impurities are provided in Table 7 and Table 8.

Table 7. Chemical Specifications of Initial and Make-up Fuel Salts

[

]

Table 8. Allowable Concentration of Selected Impurities

[

]

Integral Pumping System

The IPS performs the [ ]. Its purpose is to [ ]. The IPS consist of [

Figure 46 shows the general configuration of this design. The fuel salt pumps are [

]. Mechanical seals will be required between the pump & the turbine in the upper unit to avoid leakage of fuel salt to coolant salt loop].

Each [

].

Within the PHX, Secondary Coolant Salt circulates on the tube-side of the PHX (while fuel salt is on shell side) and transfers the heat away from fuel salt while being isolated from the highly radioactive primary fuel salt liquid. After [

].

The flow diagram (Figure 46) also shows heat exchangers HX1 & HX2. These heat exchangers are required to remove the heat added by pump 1 to the Coolant Salt & by pump 2 to the fuel salt. Note, a priming system which will be developed during Detailed Engineering will also be provided for the IPS].

Figure 46. Flow Diagram of Integral Pumping System

[

]

Heat Exchangers HX1 & HX2

Heat exchangers HX1 & HX2 are [

].

Pumps

Each [ ]. Since total fuel salt flow required for the base design is [ ]. Based on [

].

[

].

To keep the IPS [

].

[

], is now moving into the detailed engineering phase].

[

-
-
-
-].

Graphite Moderator

The purpose of the reactor moderator (also referred to as the core) is to provide the medium for slowing down neutrons to promote the nuclear chain reaction. A thermalized or softer spectrum significantly lowers the enrichment and/or fuel concentration needed for the molten fluoride fuel salt as well as providing a longer neutron lifetime which aids reactor control. Table 9 provides specific design data for the graphite moderator. Figure 47, Figure 48, and Figure 49 depict the current design of the graphite moderator. The reactor moderator is designed such that fuel channels are created which provide passages for fuel salt to flow upwards, using pump force (or natural circulation during certain accident scenarios), to the PHX.

Table 9. Graphite Moderator Design Data (R-76)

[

]

Figure 47. Core Shield Plan View

[

]

Figure 48. Core Shield Detail

[

]

Figure 49. Core Shield Section View

[

]

Table 10 below describes the main functions of the Reactor Moderator:

Table 10. Reactor Moderator Functions

Requirements	Basis
Provide a medium for slowing down neutrons to promote nuclear chain reaction	To sustain and promote nuclear fission chain reaction
Facilitate sufficient rate of primary molten fuel salt circulation through moderator channels	To enable flow circulation of primary salt required to ensure transport of heat generated and deposited in the reactor moderator during normal operation and design-basis accident events to the primary heat exchangers and/or (for the latter) to IRVACS.
Meet its functional objectives with minimum pressure drop across the graphite moderator during normal operation.	To minimize required pumping power and allow natural circulation if the pumps are unavailable.

Requirements	Basis
Provide a boundary between the central fuel area and the downcomer.	To maintain fuel circulation within the Core-unit.
Minimize the heat transfer to the fuel in the downcomer.	To promote fuel circulation.
Provide undisturbed access for the insertion of shutdown rods.	To provide a path inside the moderator core for reactivity mechanisms.
Be designed and constructed so that Fuel Salt within the moderator core assembled components do not offer any liquid fuel retention spots/pockets.	To facilitate efficient fuel utilization.

Reflector

The Reflector is a region of graphite without fuel channels which surrounds the graphite moderator. Introduced in the original reactor physics assessment the purpose of the Reflector is to return escaping neutrons to the active Core region, improving neutron economy, and to slow down escaping fast neutrons so they may be better absorbed by the thermal neutron shield, thus reducing the fast neutron flux impinging on the Downcomer and RV.

Table 11 Reflector Design Data (R-76)

[

]

Shielding

Thermal Neutron Shield

A Thermal Neutron Shield [

].

Reduction in the [

] (R-82).

Combined with the [

]

(R-82).

Table 12. Thermal Neutron Shield Design Data (R-76)

[

]

Gamma Shield

The Gamma Shield is located [

] (R-82).

Table 13. Gamma Shield Design Data (R-76)

[

]

Upper Hold-down Plate

The Upper Hold-down Plate is a [

].

The Upper Hold-down Plate has [

]. The Upper Hold-downplate also [

]. Figure 50 shows a more detailed view of the component.

Table 14. Upper Hold-down Plate Design Data (R-10)

[

]

Figure 50. Upper Hold-down Plate Detail

[

]

Lower Support and Flow Guide

The Lower Support and Flow Guide is [ ]. The Lower Support and Flow Guide provide a [ ].

Orifices in the [ ]. Figure 51 provides a more detailed view.

Table 15. Lower Support and Flow Guide Design Data (R-76)

[

]

Figure 51. Lower Support and Flow Guide Detail

[

]

Chimney

After leaving the graphite moderator, circulating fuel salt [ ]. Figure 52 shows the location of the Chimney within the Core-unit. Fuel salt [ ]. Finally, [ ] of the RV.

Table 16. Chimney Design Data (R-79)

[

]

Figure 52. Chimney Location

[

]

Downcomer Duct

After leaving the [

].

Table 17. Downcomer Duct Design Data

[

]

Internal Structural Steel

Within the RV, all the loads from [

].

However, when the filled with molten fuel salt, the [
]. Additionally, while the IPS is operational [

].

Shutdown Rods

Reactivity control for the IMSR® design consists of long-term changes to reactivity due to burnup being handled by the fuel salt makeup. Short-term transients are controlled by the inherent response of the negative temperature/power reactivity coefficient of the design. The IMSR® is designed such that an automatic reactor shutdown is not required for any AOO or DBA to reach a safe end-state (a safe end-state for IMSR® is defined to be the reactor at a 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 will include an independent means of shutting down the reactor. The IMSR400 shutdown requirements are documented in, IMSR400-30000-DG-008, Means of Shutdown (R-34).

The following is from SDM Design Description, IMSR400-22141-DD-001 (R-35). The principal functions of the SDM are:

a. to insert the negative reactivity of the Shutdown Rods into the reactor core to initiate the Guaranteed Shutdown State (GSS) at the start of a normal shutdown.
b. to maintain the reactor in the GSS state as long as required.
c. to retract the Shutdown Rods for a normal start-up of the reactor.
d. During reactor operation, to store the Shutdown Rods at a specified elevation and hold them until an insertion into the core is required.

The secondary function of the SDM is to insert the shutdown rods in response to certain measured parameters during abnormal events. The abnormal events include Anticipated Operating Occurrences (AOOs), Design Basis Accidents (DBAs), and Beyond Design Basis Accidents (BDBAs). As an example, for the Station Black Out (SBO), a BDBA, the shutdown rods could be inserted to shut down the reactor to mitigate against overheating of the reactor structures. It is emphasized that the activation of the SDM following a SBO and other abnormal events is not required for reactivity control (ensured by the inherent safety design) but serves a defense-in-depth purpose to provide investment protection for the

IMSR® Plant.

The SDM consists of the shutdown rods, [[ ].

The critical design requirement for the SDM is to provide, whenever requested, an insertion of sufficient negative reactivity into the core to maintain the reactor in a GSS (subcritical). In the IMSR® SDM, the neutron absorber in the shutdown rod is the key element that determines the shutdown rod reactivity worth and, therefore, the negative reactivity of the SDM system as a whole.

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 state as decay heat tails off. After the shutdown rods are inserted for an extended time, it is necessary to turn on heaters for the reactor core vessel to ensure that the fuel salt does not freeze. Alternatively, the fuel salt would be pumped out and maintained in an external vessel if the heaters are unavailable or a cold state is needed for other maintenance purposes.

The shutdown rods are suspended out-of-core by a motor drive mechanism. When power is cut to the mechanism, the rods will drop under the force of gravity. The I&C systems will control power to the mechanism and will allow removal of the shutdown rods via control of the rod motors, with a speed limit to be determined. It will not be possible to drive the rods out of the core until the trip relay is re-energized by resetting the trip logic.

In the IMSR® design, an [ ]. The shutdown rods are composed of neutron-absorbing material with enough negative reactivity worth to bring the reactor to a sub-critical state for the long-term. The rods can be used to facilitate normal shutdown for plant maintenance, or at the end of the Core-unit operational life to allow defueling and transfer out of the silo.

Due to the inherent and passive safety features of the IMSR®, there are currently no automatic trips that would be needed to mitigate DBAs. The manual trip and reset functions are available in both the MCR and SCA control center locations.

The SDM will be testable without requiring system activation. The manual trip is tested by initiating a single channel manual trip and confirming that the trip signal is sent via the annunciation system. The shutdown rods will not drop due to the [ ] logic. This test is repeated for [ ] channelized manual trip buttons. Another test will be via [ ] the core. The position indications will be used to confirm the partial drop distance. The test logic is designed to avoid actuating the latching function of the trip logic which would otherwise result in a full drop of the rod. When the partial rod drop test is complete, the rod will be withdrawn to its fully out position.

SDM is required to bring the reactor to a GSS for maintenance purposes or at end-of-life of the reactor core. This supports the fundamental function of "capability to shut down and maintain the reactor shut

down." Manual lock-out controls are provided for shutdown guarantee by the SDM. These controls ensure that the shutdown rods have been inserted and block removal of the rods by the operator or other means. The manual lock-out control is designed to minimize the chance of inadvertent removal by the operator.

Primary Heat Exchangers (PHX)

The Primary Heat Exchangers (PHXs) provide heat transfer between the circulating fuel salt and a separate closed–loop secondary coolant Salt. [

]. The PHXs receive the upward flowing fuel salt that has been heated by the reactor core and transfers the heat to the coolant salt, and then directs the fuel salt flow towards the downcomer annulus. The fuel salt then enters the core through the lower plenum of the bottom plate. Figure 53 provides a schematic of a primary heat exchanger. The general arrangement of the Pie-Shape PHX is shown in Figure 54.

Table 18. Primary Heat Exchanger Design Data

[

]

Figure 53. PHX Schematic

[

]

The Primary Pumps [

].

The secondary coolant salt circulates on the tube-side of the PHX. This coolant salt transfers the heat away from the reactor core while being isolated from the highly radioactive fuel salt liquid. The secondary coolant salt running through the PHXs has independent loops, with each loop having its own coolant salt pump, isolation valves and piping. The coolant lines (inlet pipes and outlet pipes) penetrate through the Core-unit vessel head and the containment boundary. The PHXs are designed to transfer the total core heat load, which is equal to the thermal power produced in the reactor core, plus some additional heat load which is added from decay heat of internally delayed fission off-gases.

The tube side flow is the secondary coolant salt with [

].

The [

].

With the Primary Pumps [
]. During operation, the fuel salt level [
].

Figure 54. PHX Detail

[

]

Core-unit Design Loads

The Design, Service, and Test Loadings, for the RV pressure boundary components and supports, will be identified considering all plant or system operating and test conditions anticipated or postulated to occur during the intended service life of the component or support as required by the [

The Core-unit loads include but are not limited to the following:

1. [
2.
3.
4.
5.

6.

7.]

Load Combinations

[

].

Acceptance Criteria

[

]

Control and Instrumentation in the Core-unit

Table 19 provides a summary of the parameters which will be measured and for what purpose for SSCs of the Integral Core-unit. These values are the expected measurements and control functions adapted from the Instrumentation and Controls list (R-87). An evaluation of process instrumentation for the Integral Core-unit environment has been complete (R-88) and will guide instrument selection. The exact type, location, and quantityof instruments achieving the functions in Table 20 will be finalized during Detailed Engineering based on results from the Instrumentation and Controls System Test Plan (R-89).

Table 19 Core-unit SSC Instrumentation and Control List

[

]

Core-unit Externals

Guard Vessel

A GV surrounds the Core-unit and provides defense in depth protection in case of a BDBA accident resulting in a RV leak. The GV will be constructed of stainless steel and and will be supported by the Reactor Support Structure (RSS) through the RV. Sizing of the GV will be [

].

Heaters and Insulation

Heaters will [

].

System Boundaries and Interfaces

The following systems have interfaces with the Core-unit and either provide support to or are supported by the Core-unit:

1. Containment Systems

2. Internal Reactor Vessel Auxiliary Cooling System (IRVACS)
3. Primary Reactor Auxiliary Cooling System (PRACS)
4. Defueled Core-unit Cooling System (DCCS)
5. Upper Plenum Active Cooling System (UPACS)
6. Shutdown Mechanism (SDM)
7. Reactor Support Structure (RSS)
8. Irradiated Fuel System (IrFS)
9. Makeup Fuel System (MFS)
10. Initial Fuel System (InFS)
11. Gas Management System (GMS)
12. Secondary Coolant System (SCS)
13. Core-swap System
14. Electrical Power Systems
15. Plant Control and Monitoring System (PCMS)
16. Reactor Auxiliary Building (RAB)
17. Instrumentation & Control System

VIII. 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 Reactor Vaults; this accommodates switching to a new Core-unit every seven years. One Reactor Vault is for the operating Core-unit, and one Reactor Vault 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:

Table 20. Core-unit Replacement Sequence

[

]

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

Start-up

Start-up can occur following one of three shutdown states: hot, warm, or cold shutdown. These states are defined by the duration of shutdown which will be a function for the reason behind the shutdown. Start-up under NO will include the entire range of start-up from these different types of shutdowns. However, not included here is the initial start-up of the first Core-unit. Initial fueling (through the Initial Fueling System) and start-up are considered commissioning activities, and thus not part of NO, as they are only performed once in the lifetime of the reactor.

The process for the reactor to start-up (i.e., to achieve criticality) will be assessed during the basic and

Detailed Engineering project phases. The preferred option is to fill the Core-unit with fuel salt (if it had previously been defueled) while the SDM rods are inserted, heat the fueled Core-unit to above the criticality temperature with the SDM rods still inserted, remove the SDM rods, then let the reactor cool to become critical.

Start-up procedures will include a method for measuring how subcritical the core is and to estimate how long it will take to make the reactor critical. The following is the conceptual operational sequence:

a. [

b.

c.

d.

e.

f.

g.

h.

  i.

] mode.

A detailed start-up procedure will be developed through additional physics and thermalhydraulics analysis during Detailed Engineering.

Low-Power and Critical

After start-up, power may be ramped and held at low power. This may be done for a fewreasons, including to conduct some systems testing. The following is the conceptual operational sequence:

j. [

k.

l.

m.]

High-Power/Full-Power

Usually following criticality, reactor power is ramped into the high-power/full-power range to allow electrical power generation. The following is the conceptual operational sequence:

1. [

2.

3.].

Figure 55 shows an example of an operator's mental model of the IMSR400 main systems during Normal Operations: High-power.

Figure 55. Normal Operations: High-power

[

]

Addition of Positive Reactivity

Make-up fuel salt additions provide the long-term burnup compensation (i.e., compensate for the depletion of fissionable isotopes in the fuel salt). The following is the conceptual operational sequence:

1. [
2.
3.
4.
5.]

Core-swap

Each RAB contains one active Core-unit. Due to the 7-year design life of each Core-unit, a process must be implemented to change the active Core-unit over the life of the IMSR400 facility. The process is

facilitated by the Core-swap System (R-90).

The first Core-unit will be installed in Reactor Vault A and started up with initial fuel salt. After the first Core-unit 7-year life is over, it is shut down and its Irradiated fuel salt is transferred to the FSST. A new Core-unit is installed in Reactor Vault B and started up with irradiated fuel salt from the FSST.

After an additional approximately 7-years, the first Core-unit has cooled and may be removed from Reactor Vault A and placed in a Used Core Storage Silo within the RAB. The Core-swap system features Robotic Manipulators which disconnect the used Core- unit piping and instrumentation from the vault. The Large Crane is used to move the Core- unit to the silo. The disconnection and removal of a Core-unit is a function of the Core- swap System and is described in further detail in (R-90). A new Core-unit is installed in the Reactor Vault A from which Core-unit one was removed by the Core-swap System.

The second Core-unit is shut down and defueled to the FSST where it's irradiated fuel salt mixes with the excess load from the first Core-unit. The third Core-unit is started up from the mixed irradiated fuel salt in the FSST, as well as possibly an initial quantity of makeup fuel salt to account for decay within the FSST. This process continues for the life of the IMSR400, with the Active Core-unit alternating between Reactor Vault A and B. Used Core-units will be moved to Used Core Storage Silos, except for the final two Core- units which will remain in place.

IX. Conclusion

In accordance with TEUSA's Regulatory Engagement Plan (REP), the Company intends to submit an application for an SDA of the IMSR® Core-unit consistent with the requirements established in 10 CFR 52, Subpart E. To help establish the basis for the SDA application, TEUSA has defined the IMSR® Core-unit in this white paper by providing an overview of the major SSCs that make up the overall IMSR® plant and clarifying which are included in the IMSR® Core-unit and by exclusion, those that are not.

This white paper supports the IMSR® Core-unit SDA application development efforts. As outlined in the REP, additional technical documents are being developed and will be submitted to the NRC.

This white paper will be periodically updated to reflect the continuing evolution of the design. The subsequent topical reports that will be submitted on a variety of technical topics and system designs will rely on the content of this report to provide the basic information needed to support the NRC of the submitted topical reports. Each subsequent topical report may also include additional information unique to the subject review that would be necessary to provide a sufficient technical basis for approval of the topical report.

X. Abbreviations & Acronyms

AC – Alternating Current

ALARA – As Low As Reasonably Achievable

AOO – Anticipated Operational Occurrence

BeF4 – Beryllium Fluoride

BFP – Boiler Feed Pumps

BDBA – Beyond Design Basis Accidents

CAAS – Criticality Accident Alarm System

CB – Control Building

CFD – Computational Fluid Dynamics

CFR – Code of Federal Regulations

DBA – Design Basis Accidents

DC – Direct Current

DEC – Design Extension Conditions (Beyond Design Basis Accident)

DMSR – Denatured Molten Salt Reactor

EHX – Emergency Heat Exchanger

EQ – Environmental Qualification

FACS – FSST Active Cooling System

FSS – FSST Support Structure

FSST – Fuel Salt Storage Tank

FVCS – FSST Vault Cooling System

GHTCS – Gas Holding Tank Cooling System

GV – Guard Vessel

HCCS – Hot Cell Cooling System

HEPA – High Efficiency Particulate Air

HVAC – Heating, Ventilation, and Air Conditioning

HX – Heat Exchanger

I&C – Instrumentation and Control

I/O – Input/Output

I-NPP – IMSR® Nuclear Power Plant

IFACS – Irradiated Fuel Auxiliary Cooling System

IFCS – Irradiated Fuel Cooling System

ILW – Intermediate Level Waste

IMSR – integral Molten Salt Reactor

IPS – Integral Pumping System

InFS – Initial Fuel System

IrFS – Irradiated Fuel System

IRVACS – Internal Reactor Vessel Auxiliary Cooling System

KF – Potassium Fluoride

LEU – Low Enriched Uranium

LiF – Lithium Fluoride

LLW – Low Level Waste

M – Meters

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

NCPS – Nuclear Plant Control System

OCC – Out-of-Core Criticality

ORNL – Oak Ridge National Laboratory

PCMS – Plant Control and Monitoring System

PDC – Principal Design Criteria

PDS – Plant Display System

PIE – Postulated Initiating Events

PHX – Primary Heat Exchanger

PRACS – Process Reactor Auxiliary Cooling System

PSA – Probabilistic Safety Assessment

RV – Reactor Vessel

R&D – Research and Development

RAB – Reactor Auxiliary Building

REP – Regulatory Engagement Plan

RSS – Reactor Support Structure

SC – Safety Classification

SCA – Secondary Control Area

SCS – Secondary Coolant System

SCS (instrumentation) – Secondary Control System

SDA – Standard Design Approval

SDM – Shutdown Mechanism

SGB – Steam Generation Building

SHX – Secondary Heat Exchanger

SMS – Secondary Monitoring System

SPCS – Steam Plant Control System

SQ – Seismic Qualification

Sr - Strontium

SS – Stainless Steel

SST – System Service Transformer

TB – Turbine Building

TCS – Tertiary Coolant System

TEI – Terrestrial Energy, Inc.

TEUSA – Terrestrial Energy USA, Inc.

TSV – Thermal Static Valve

U. S. – United States

U – Uranium

UCE – Upper Containment Enclosure

UCH – Upper Containment Head

UF4 – Uranium Tetrafluoride

UST – Unit Service Transformer

VDR – Vendor Design Review

VSD – Variable Speed Drive

Xe - Xenon

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