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Postulated Initiating Events for the IMSR®

Abstract

This white paper describes the process used in developing a set of postulated initiating events (PIE) for the Integral Molten Salt Reactor (IMSR®) design and provides as complete a set of PIE as is possible given the design's maturity. The set of PIE are not confined to the events that only affect the IMSR® Core-unit. Instead, the set of PIE discussed in this white paper, and associated references, result from an examination of the complete IMSR® facility. This examination includes the potential off-normal conditions that can lead to potential challenges to one or more of the fundamental safety functions.

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I. Purpose

The purpose of this white paper is to describe the process used in developing a set of postulated initiating events (PIE) for the Integral Molten Salt Reactor (IMSR®) design, and to provide as complete a set of PIE as is possible given the design's maturity. The set of PIE discussed in this white paper are not confined to the events that only affect the IMSR® Core-unit. Instead, the set of PIE discussed in this white paper and associated references result from an examination of the IMSR® facility as a whole. The examination includes the potential off-normal conditions that can lead to potential challenges to one or more of the following fundamental safety functions:

- Controlling reactivity
- Removing heat from the primary system
- Retaining radionuclides

II. Introduction

Licensing Strategy and Objective

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

TEUSA has developed and submitted a Regulatory Engagement Plan (REP) (Reference 1) to the Nuclear Regulatory Commission (NRC). The REP outlines a series of technical documents which TEUSA plans to submit to the NRC for review and comment or review and approval. Consistent with the REP, TEUSA has submitted several white papers that have 1) provided an overview of major plant buildings, structures, systems and components that make up the IMSR® facility, 2) identified the interfacing systems that provide important functions in support of the operation and safety of the IMSR® facility and that interact directly with the IMSR® Core-unit, and 3) have provided a proposed set of principal design criteria (PDC) that will be used for the IMSR® facility. The TEUSA REP (Reference 1) provides additional details regarding TEUSA's licensing activities and objectives.

The Identification of PIEs is the first step in the design review and assessment process. It is also a starting point for the safety analysis. A PIE is an event identified in the design that leads to Anticipated Operational Occurrences (AOOs) or accident conditions. Accident conditions include Design Basis Accidents (DBAs) and Beyond Design Basis Accidents (BDBAs). BDBAs include a subset of severe accidents.

The main objective of this report is to identify all foreseeable PIEs, group them into categories, and classify the PIEs at this stage based on engineering judgement into AOOs, DBAs, and BDBAs.

There are several sources used to identify the PIEs. The basic sources include Probabilistic Safety Assessment (PSA) results, engineering judgement, reference lists for the same or similar type of reactor, and operational experience. The operational experience from the Molten Salt Reactor Experiment (MSRE) contains some actual AOOs and experiments which provide insight on DBAs. Therefore, in the current work, the PIEs of the IMSR400 are identified based on a systematic approach supplemented by MSRE experience (in some cases) and engineering experience. The PIEs are further grouped into categories, based on similarity of the initiating failures, key phenomena, or system and operator response. The resulting list of PIEs may need to be adjusted as the design evolves. An IMSR400 PSA is presently under development. The final set of PIEs will be finalized and the final classification of PIEs will be confirmed or will be adjusted at the time the results of the PSA become available.

III. Scope

The scope of the PIE assessment considers a single unit nuclear power plant configuration. The business plan for the IMSR400 is evolving towards a two-unit per site configuration. A final PIE assessment will be performed once the multi-plant configuration and the PSA are completed. The PIE assessment includes plant operating states of startup, at-power, and shutdown (including core-swap). The PIE assessment examined major structures, systems, and components (SSCs) including the Reactor Vessel and its contents (i.e., Core-unit, primary pumps, primary heat exchangers, and fuel salt), secondary and tertiary cooling systems (including the secondary cooling system heat exchangers), the steam generator heat exchangers and any other SSCs whose failure or malfunction could negatively impact upon operation of the nuclear power plant, the irradiated fuel system (IrFS), the initial and makeup fuel systems (InFS and MFS), the Core swap system, graphite core support structures, and the Internal Reactor Vessel Auxiliary Cooling System (IRVACS). At this time, the final design of the IRVACS is being revised. As such, this white paper version will not reflect the pending modifications of the IRVACS.

The scope of the assessment also includes PIEs that impact the nuclear power plant as a whole, such as the loss of offsite power, and internal or external common cause events. Potential external events include environmental events (flooding, fire, earthquake, etc.), malicious attacks (electromagnetic interference and cyber-attacks), and internal system failures (explosions or turbine missiles).

IV. Process for Identifying PIE

In the design stage of IMSR400, it is important to identify a comprehensive set of PIEs such that, to the extent practical, all foreseeable events are identified and are considered in the design. From the perspective of deterministic safety analysis, a comprehensive listing of PIEs should be prepared for all permissible plant operating modes (reactor startup, at-power and shutdown) to ensure that the analysis of plant behavior is complete. After the list of PIEs is generated for safety assessment studies, the events are grouped into different categories i.e., AOO, DBA and BDBA.

The list of PIEs is prepared in this white paper were prepared to ensure that the analysis of behavior of the IMSR400 is complete. To achieve a list of PIE that is as complete as possible, the identification process used was as systematic and complete to the greatest practical extent. The various causes for an initiating event were considered, which include operator errors and equipment failures, human induced or natural events, and internal or external common cause events that directly or indirectly challenge one or more of the systems required to maintain the safety of the plant.

In this white paper, the highlights of the "top-down" approach are presented. A more detailed discussion is presented in Reference 2. The top-down approach starts by [

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As mentioned above, an undesired outcome should be specified first. For IMSR400, the top event is taken as [

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These steps are reflected in the following referenced figures, which are wholly proprietary and are included in Reference 2 as logical steps in the development of potential accident initiating events. The references to the figures are provided to assist the staff in auditing the development process. Figure 1 shows the first several levels of the top-down flowchart. Figure 2 lists the possible external common cause events and internal common cause events that could potentially cause radioactive release. The internal events causing radioactive release are further divided into two categories: events inside the Reactor Vessel and events outside the Reactor Vessel. Most radioactive materials are in the Fuel Salt, moderator, Fuel Salt Storage Tank (FSST), and Irradiated Fueling System (IrFS). Therefore, it is necessary to find out how these materials could become mobile, i.e., airborne or water borne. Radioactive material could be released from the fuel and other systems by mechanical damage (tank and pipe leakage) or boundary failure from excess temperature or pressure. Figure 3 shows the identification flowchart for initiating events outside of the Reactor Vessel. Figure 4 shows the initiating events caused by mechanical boundary failure inside the Reactor Vessel. Figure 5 displays the identification flowchart for initiating events caused by positive reactivity insertion inside the Reactor Vessel.

One of the main potential causes of boundary failure is loss of heat removal. Figure 6 shows the identification flowchart for initiating events due to loss of heat removal inside the Reactor Vessel. Figure 7 shows the identification flowchart for initiating events due to loss of heat removal outside the Reactor Vessel.

At this stage of the IMSR® design maturity, TEUSA has projected that most of the PIEs will have minimal or no consequences because of the fundamental nature of design and passive/inherent safety features of the IMSR400 facility. A more detailed radiological consequence assessment will be performed after the PSA is completed, which will occur prior to TEUSA submitting an SDA application.

V. Criteria used for Categorizing Postulated Initiating Events.

In order to place PIEs into accident categories, TEUSA will be using the following qualitative criteria and some quantitative criteria. TEUSA notes that the category ranges are consistent with event classification efforts in numerous industry and NRC documents. The first category is anticipated operational occurrences. This category is expected to capture all postulated events that might be expected to occur one or more times in the lifetime of a plant. Using an assumed lifetime of 60 years (the current design basis for the IMSR400), an estimated frequency for AOOs will be down to a frequency likelihood range per event of 10^{-2}/reactor year. The next category of events is design basis events. These events are much less likely to occur and would generally capture events that are expected to occur once in the lifetime of a large population of plants. If one assumes a population of 1000 reactors, events in this category would have frequency likelihood events ranging from 10^{-2} through 10^{-5}/reactor year. Any events with a frequency likelihood less than 10^{-5} through $5x10^{-7}$ would be considered beyond design basis events. Events with a frequency likelihood less than $5x10^{-7}$ are considered so rare that they would not need to be considered when assessing the capabilities of the plant to respond to those postulated events.

Figure 1 – Main PIEs Identification Top-Down Flowchart of IMSR400 Radioactive Release

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Figure 2 – PIEs Identification Top-Down Flowchart of Internal and External Common Cause Events

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Figure 3 – PIEs Identification Top-Down Flowchart for Events from Outside of RV

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Figure 4 – PIEs Identification Top-Down Flowchart for Events from Mechanical Boundary Failure of Inside of RV

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Figure 5 – PIEs Identification Top-Down Flowchart for Events from Reactivity Insertion

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Figure 6 – PIEs Identification Top-Down Flowchart for Events from Loss of Heat Removal Inside of RV

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Figure 7 – PIEs Identification Top-Down Flowchart for Events from Loss of Heat Removal Outside of RV

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VI. IMSR® Power Plant Description - Overview

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

The IMSR® Safety Case is the most powerful driver of plant cost-innovation as it exemplifies the nuclear industry's "Control, Cool, Contain" safety framework. The simplicity, security, and robustness of the IMSR® design result from the immutable properties of fuel, materials, and from the simple architecture of the plant and equipment. This extensive use of such an architecture to achieve passive safety mechanisms is fundamentally not possible with solid fueled reactor systems. The result is significant cost reductions coming from a simplified design, less onerous design requirements, reduced material cost, and reduced scope of regulatory approvals.

Reactor Profile

Reactor Type: The IMSR® is a thermal spectrum, pool-type, molten fluoride salt fueled reactor.

Reactor Coolant: The IMSR® is fueled and cooled by a molten fluoride salt compound containing standard assay low enriched uranium (LEU). The initial fuel load enrichment is expected to be ~2% enriched ^{235}U, and the makeup fuel is expected to be 4.95% enriched ^{235}U.

Reactor Product: 600 °C heat in the form of a molten solar salt to be used for electricity production, industrial heat applications, or a combination of both.

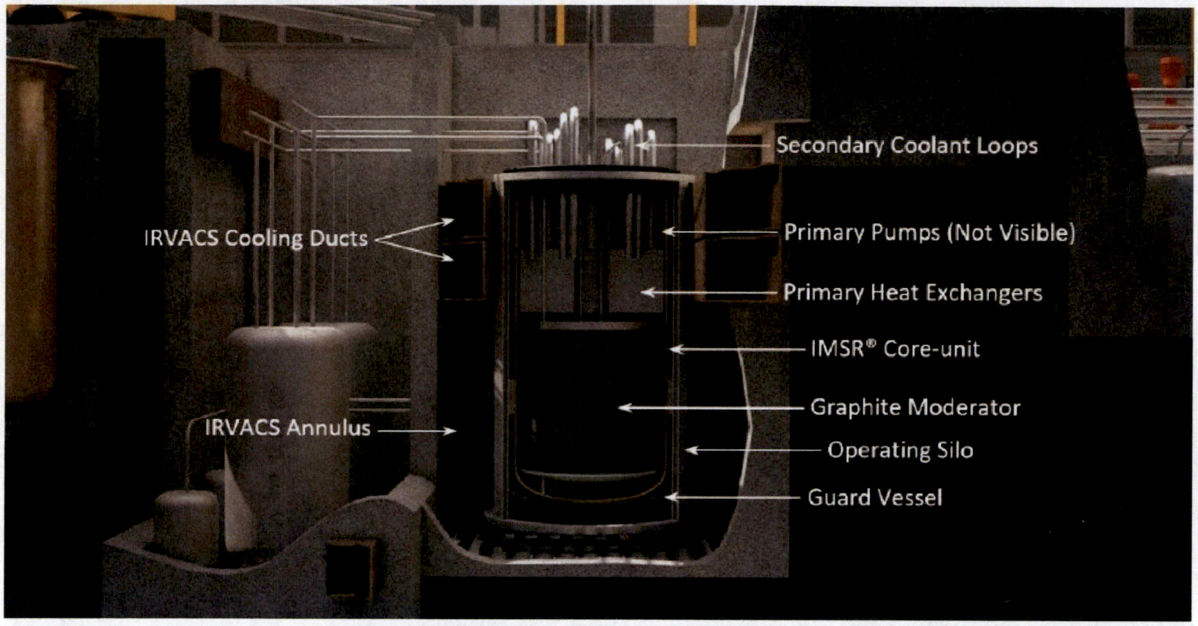
Power Output: 442 MWth (195 MWe)

Thermal Efficiency: 44% (net)

Reactor and Power Conversion Process

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 fissioning of the fuel raises the temperature of the fuel salt as it passes through the moderator region. The fuel salt then exits the graphite moderator and is transferred to the Primary Heat Exchangers.

Figure 8: IMSR® Core-unit and Surrounding Structures, Systems and Components



Heat is transferred from the Core-unit via a secondary coolant system, a system using non-radioactive molten salt as the coolant, which passes through the Primary Heat Exchangers. The secondary coolant salt then passes through the secondary heat exchanger, where heat is transferred to a tertiary molten salt coolant loop. After passing through the secondary heat exchanger, the now cooler secondary coolant salt then returns to the Primary Heat Exchanger within the IMSR® Core-unit. The Secondary Coolant Loop connections, Primary Pumps and other components associated with the Core-unit is shown in Figure 8 above.

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

Site Layout

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. As mentioned earlier, the IMSR400 site layout is evolving to accommodate a two-unit configuration. The site changes needed for the two-unit configuration will be described in the next revision to this document, however, the major buildings and systems for each unit will not be radically changed from the information presented below. A typical site has a small footprint (about 7 hectares or 17 acres) and a small security perimeter (approximately 130m x 145m).

An IMSR® Nuclear Power Plant (I-NPP) site includes a Reactor Auxiliary Building, Turbine Building, Control Building, and Maintenance Building. Also included are the plant support buildings and structures. These include the Administration Building, Simulator and Training Building, Radioactive Waste Building, Coolant Salt Storage Building, Emergency Mitigating Equipment Building, Main Pump

House, Water Treatment Building, Fire Water Pump Building, Cooling Water Outlet Building, Sewage Treatment Plant, Electrical Switchyard, and a Security Building.

Figure 9: Typical Site Layout for a Single I-NPP

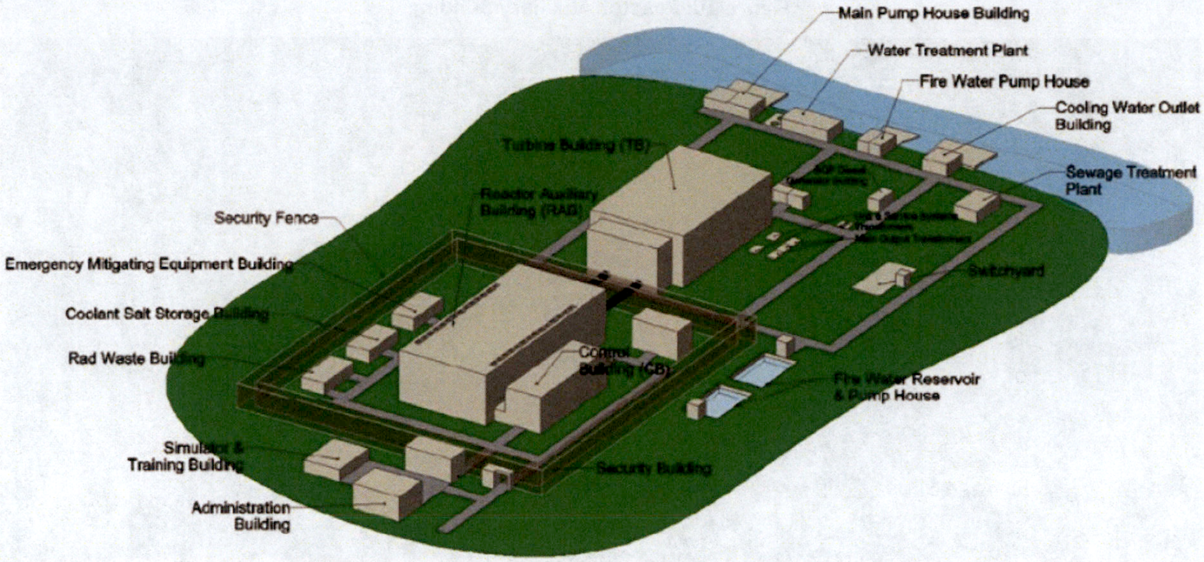


Figure 9 above represents a simplified plant layout identifying the arrangement and configuration of the major buildings, structures, and boundaries of an I-NPP site. A "generic design site envelope," is used to develop the IMSR® site design and encompasses generic site parameters used in Canada, the U.S. and European countries relevant to nuclear plant siting.

Structures, Systems and Components that Comprise the IMSR®

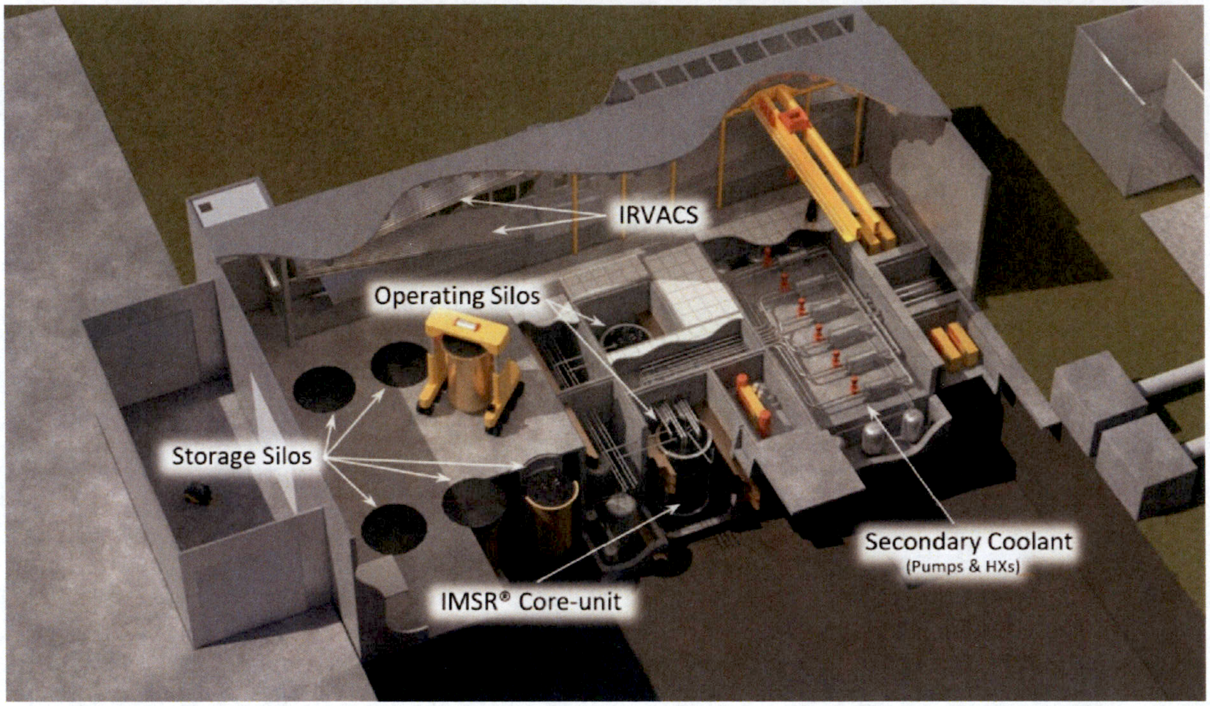
The following list identifies the significant structures, systems, and components (SSC) associated with the IMSR® facility.

1. Reactor Vessel
2. Guard Vessel
3. Liquid Fuel Salt
4. Primary Pumping System
5. Graphite Moderator
6. Shutdown Rods
7. Primary Heat Exchangers
8. Instrumentation and Control
9. Makeup Fuel System (MFS)
10. Irradiated Fuel System (IrFS)
11. Secondary Coolant System
12. Internal Reactor Vessel Auxiliary Cooling System (IRVACS)
13. Main Control Room and Secondary Control Area
14. Silos
15. Reactor Support Structure

16. Containment
17. Reactor Auxiliary Building

A general description of each of these SSC's is provided in the TEUSA white paper titled "*Definition of the IMSR Core-unit*" submitted to the NRC in March 2020 (Reference 3).

Figure 10: Reactor Auxiliary Building



VII. Summary of PIE results

The PIEs for IMSR400 are identified based on the top-down flowcharts presented in Section V. Currently, TEUSA has identified a total of 104 internal initiating events and possible common cause events (internal and external). The identified events have been grouped into categories based on similarity of the initiating failures, key phenomena, or system and operator response. In the current work, the categories used in PIE grouping are the following:

1. Unwanted Changes in Reactivity (9)
2. Increase of Core Heat Removal (6)
3. Decrease of Core Heat Removal (37)
4. Loss of Integrity of Piping or Vessel (36)
5. Internal Common Cause Event (4)
6. External Common Cause Event (9)
7. Additional Events (5)

Each of the categories of events will be discussed in the following sections. Note that the set of internal and external common cause events listed in this document is a typical list based on information in Reference 7.

VIII. PIE List

TEUSA has developed a complete PIE list with a brief description and expected related mitigation method (to be confirmed by the safety analysis) for each initiating event, as shown in this section. In this list, only initiating events are shown. Combination of events are not included in this report but will be included as part of the PSA. For the IMSR400, shutdown is listed as a mitigation method when it is expected to occur; the detailed design will indicate the appropriate parameters. Safety analysis will not credit engineered shutdown in any case, however, engineered shutdown may be credited as a defense in depth feature in the PSA. The PIEs listed do not credit heat removal via the process heat sink (i.e., heat removal via the secondary and tertiary cooling systems and the steam conversion system).

The issue of core ageing is important but is not an initiating event. Therefore, issues related to core aging are not discussed in this white paper. Generally, once it is known how the core ages, the safety analysis of PIEs will account for this by assuming the appropriate pessimistic graphite conditions.

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IX. Classification of the PIEs

After all PIEs are identified, they are classified into the following three classes of events based on the results of probabilistic studies and engineering judgement:

- AOO: these include all events with frequencies of occurrence equal to or greater than 10^{-2} per reactor year.
- DBA: these include events with frequencies of occurrence equal to or greater than 10^{-5} per reactor year, but less than 10^{-2} per reactor year.
- BDBA: these include events with frequencies of occurrence less than 10^{-5} per reactor year.

The criteria used for the classification has been presented earlier in this report.

At the current stage, the probabilistic studies are not yet completed and so the classification shown in this document is based on engineering judgement and previous similarity with other potential events. TEUSA intends to revisit the final list of PIE and their categorization when the PSA is more mature. Any modifications or additions to the current list will be provided either as an update to this white paper or will be discussed as part of TEUSA's planned SDA application.

The engineering judgements used in this report are the following:

- For events that are similar to those in existing power reactors (e.g. loss of offsite power, pipe break, vessel leak, heat exchanger tube leak, fueling error), the same classification is followed. For events which are not similar, the following process is applied.
    - Because AOOs are reasonably expected to happen during the lifetime of a plant, all failures of plant operations system are classified as AOOs. An AOO is initiated by a single equipment malfunction, or a single operator error.
    - DBAs are the events beyond AOOs, initiated by major equipment malfunctions with no credit for operator action. Also, single serious operator errors such as a fueling error would be a DBA. Any large boundary failure other than the Reactor Vessel would be a DBA.
    - BDBAs are the events beyond DBAs, which might result in a significant release of radioactivity. For example, station blackout will be a BDBA (it is considered as a combination event and not listed in PIEs List). We are currently developing a case to demonstrate that spontaneous loss of Reactor Vessel integrity (leak) is a BDBA.

Based on the above judgement, the current classification of PIEs is listed below. This list is identical to the list of PIEs provided in Reference 2:

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X. Conclusion

This report identifies the PIEs which can potentially challenge the safety functions of the IMSR® facility and release radioactive material to the environment, and groups them into 7 categories for the future use of deterministic safety analysis (DSA) and Probabilistic Safety Assessment (PSA). Overall, the number of current PIEs considered for the IMSR400 is 104 events.

The summary of the changes to the PIE list from the previous version of the white paper are presented below:

1. Unwanted Changes in Reactivity (9) – increased by one event.
2. Increase of Core Heat Removal (6) – unchanged.
3. Decrease of Core Heat Removal (37) – increased by ten events.
4. Loss of Integrity of Piping or Vessel (36) – increased by twelve events.
5. Internal Common Cause Event (4) – unchanged.
6. External Common Cause Event (9) – unchanged.
7. Additional Events (4) – decreased by three events.

All identified PIEs are also classified into three classes (AOOs, DBAs, and DBDAs) according to the guidance from Reference 7. As of this revision, there are 29 events included as AOOs, 78 events included as design basis accidents, and 38 events that are considered to be beyond design basis events. The classification of the PIEs is based on similarity to existing NPPs and is supplemented by engineering judgement and the guidance from Reference 7. The classification needs to be further confirmed by the results of the PSA.

The list of PIEs is as complete as possible based on the current design and may change as the IMSR® design evolves through its final stage. Therefore, certain events may be added or removed from the list as needed.

XI. Abbreviations and Acronyms

AOO – Anticipated Operational Occurrence

ARDC – Advanced Reactor Design Criteria

ARE – Aircraft Reactor Experiment

BDBA – Beyond Design-Basis Accident

BeF_2 – Beryllium Fluoride

CFR – Code of Federal Regulations

CNSC – Canadian Nuclear Safety Commission

Cs - Cesium

DBA – Design Basis Accident

DSA – Deterministic Safety Analysis

DMSR – Denatured Molten Salt Reactor

EHX – Emergency Heat Exchanger

FMEA – Failure Modes and Effect Analysis

FSST – Fuel Salt Storage Tank

HX – Heat Exchanger

I&C – Instrumentation and Control

IMSR – Integral Molten Salt Reactor

InFS – Initial Fuel System

IrFS – Irradiated Fuel System

IRVACS – Internal Reactor Vessel Auxiliary Cooling System

I-NPP – IMSR Nuclear Power Plant

KF – Potassium Fluoride

LEU – Low Enriched Uranium

LiF – Lithium Fluoride

MCR – Main Control Room

MFS – Makeup Fuel System

MLD – Master Logic Diagram

MSLB – Main Steam Line Break

MW – Megawatt

MWe – Megawatt Electric

MWth – Megawatt Thermal

MSR – Molten Salt Reactor

MSRE – Molten Salt Reactor Experiment

NaF – Sodium Fluoride

NIA – Nuclear Innovation Alliance

NRC – Nuclear Regulatory Commission

NSSS – Nuclear Steam Supply Systems

ORNL – Oak Ridge National Laboratory

PCMS – Plant Control and Monitoring System

PDC – Principal Design Criteria

PHX – Primary Heat Exchanger

PIE – Postulated Initiating Event

PSA – Probabilistic Safety Assessment

QA – Quality Assurance

R&D – Research and Development

RAB – Reactor Auxiliary Building

REP – Regulatory Engagement Plan

SBO – Station Blackout

SCA – Secondary Control Area

SCS – Secondary Coolant System

SDA – Standard Design Approval

SDM – Shutdown Mechanism

SG – Steam Generator

SHX – Secondary Heat Exchanger

Sr - Strontium

SS – Stainless Steel

SSC – Structures, Systems, and Components

TCS – Tertiary Coolant System

TEI – Terrestrial Energy, Inc.

TEUSA – Terrestrial Energy USA, Inc.

U.S. – United States

VDR – Vendor Design Review

Xe – Xenon

XII. References

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