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University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Event Sequence Identification and SSC Safety Classification Methodology

TOPICAL REPORT

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EXECUTIVE SUMMARY

The University of Illinois Urbana-Champaign (UIUC) is proposing to construct a research reactor (also known as a non-power utilization facility) using high temperature gas-cooled reactor (HTGR) technology. UIUC plans to build and operate Ultra Safe Nuclear Corporation's (USNC) Micro Modular Reactor^{TM} (MMR^{TM}) HTGR based design.

This topical report (TR) describes the following processes to be used for the licensing of the MMR at UIUC:

1. The identification of credible event sequences that will be considered for the MMR design, and
2. The safety classification of systems, structures, and components (SSCs) appropriate for their function(s) in meeting the design basis.

Details of how the design addresses these aspects will be provided in the preliminary and final safety analysis reports (PSAR and FSAR, respectively).

This TR presents a simplified, deterministic approach to identify event sequences and classify SSCs that aligns with the guidance provided in NUREG-1537 (Reference 1). The standard MMR design developed by USNC uses probabilistic risk assessment (PRA) insights to inform the generic plant design; however, the UIUC deployment will not use PRA in the licensing basis and will assess design acceptability with regard to the radiological limits applicable for research reactors.

The methodologies presented in this TR are iterative and will be used numerous times as the plant design matures.

UIUC is requesting NRC review and approval of the event sequence methodology in Section 2.0 and the safety classification methodology in Section 3.0 used for the MMR at UIUC. Preliminary lists of Postulated Initiating Events (PIEs) and structures, systems and components (SSC) Safety Classifications (Appendix A and B, respectively) and the pipe breach classification example in Section 3.0 are provided for information only, NRC approval is not requested.

CONTENTS

ABBREVIATIONS & ACRONYMS

This list contains the abbreviations and acronyms used in this document.

Abbreviation or Acronym	Definition
AP	Adjacent Plant
ASME	American Society of Mechanical Engineers
CNSC	Canadian Nuclear Safety Commission
CP	Construction Permit [per 10 CFR 50]
CPA	Construction Permit Application [per 10 CFR 50]
DID	Defense-in-Depth
D-LOFC	Depressurized loss of flow cooling
ECCS	Emergency core cooling system
EPRI	Electric Power Research Institute
FCM™	Fully Ceramic Micro-Encapsulated
FMEA	Failure Modes and Effects Analysis
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
HALEU	High Assay Low-Enriched Uranium
HPB	Helium Pressure Boundary
HTGR	High Temperature Gas-Cooled Reactor
HTS	Heat Transport System
HVAC	Heating, Ventilation, and Air Conditioning
IAEA	International Atomic Energy Agency
I&C	Instrumentation and Controls
IHX	Intermediate Heat Exchanger
LWR	Light Water Reactor
MHA	Maximum Hypothetical Accident
MHTGR	Modular High Temperature Gas Reactor
MLD	Master Logic Diagram
MMR™	Micro Modular Reactor™
MSS	Molten Salt System
MW	Megawatts
NEIMA	Nuclear Energy Innovation and Modernization Act [115-439 (01/14/2019)]
NP	Nuclear Plant
NRC	[U.S.] Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Document
OL	Operating License [in accordance with 10 CFR 50]
OLA	Operating License Application [in accordance with 10 CFR 50]
PDC	Principal Design Criteria
P&ID	Piping and Instrumentation Diagram
PIE	Postulated Initiating Event
P-LOFC	Pressurized Loss of Flow Cooling
PRA	Probabilistic Risk Assessment
PSAR	Preliminary Safety Analysis Report
PSE	Planned Special Exposure

Abbreviation or Acronym	Definition
PWR	Pressurized Water Reactor
RCCS	Reactor Cavity Cooling System
RCSS	Reactor Control and Shutdown System
RPS	Reactor Protection System
SPDS	Safety Parameter Display System
SSCs	Systems, Structures, and Components
TEDE	Total Effective Dose Equivalent
TR	Topical Report
TRISO	Tri-Structural Isotropic
UCO	Uranium Oxycarbide
UIUC	University of Illinois Urbana Champaign
U.S.	United States
USNC	Ultra Safe Nuclear Corporation (i.e., the reactor design vendor)

1.0 INTRODUCTION

USNC has undertaken a project to build an MMR at UIUC to serve as a research reactor (i.e., a Class 104 utilization facility) in accordance with 10 CFR 50.21(c).

1.1. PURPOSE

The purpose of this TR is to describe the UIUC MMR methodologies used for:

1. The identification of credible event sequences that will be considered for the MMR design, and
2. The safety classification of SSCs appropriate for their function(s) in meeting the design basis.

A definition for the term event sequence is provided in Section 2.2.

1.2. UIUC MMR DEPLOYMENT BACKGROUND

The MMR is a non-power research HTGR that will be licensed to a maximum operating power limit of 10 MW_{th} for deployment to the UIUC. The MMR is an HTGR limited to the research reactor power limit for the UIUC deployment. The reactor will be fueled with High Assay Low-Enriched Uranium (HALEU) at an enrichment from 5% to 19.75% ^{235}U in the form of tri-structural isotropic (TRISO) particles embedded in silicon carbide Fully Ceramic Micro-Encapsulated (FCM™) pellets that are stacked in columns in solid hexagonal graphite blocks. The MMR is designed for passive safety response to event sequences in the design basis and relies on functional containment as the primary means to limit release of radioactivity to the environment.

UIUC will apply for a Construction Permit (CP) per 10 CFR 50 per the schedule provided in Reference 13. As a non-water, non-power reactor, the UIUC MMR does not match the underlying assumptions that form the basis for many NRC regulations (as discussed and assessed in Reference 4).

In Reference 3, the NRC discusses approaches to improving the timeliness and efficiency of advanced reactor licensing reviews through early interactions. A key action is submitting TRs for staff review and approval. One such TR should discuss the “proposed process for selection of licensing basis events and classification and treatment of SSCs…”.

1.3. MMR TECHNOLOGY BACKGROUND

The MMR uses technology and safety capabilities considerably different from Light Water Reactor (LWR) technology that is the focus of many regulations. For example, the MMR does not require an active or passive emergency core cooling system (ECCS) to rapidly replenish primary coolant to recover the fuel in the event of a rupture of the primary pressure boundary. Large safety margins are provided by both the fuel and the reactor design.

- The fuel is comprised of TRISO particles, which provide a highly effective fission product retention capability. In response to an Electric Power Research Institute (EPRI) TR on the performance of TRISO fuel, the U.S. NRC issued a Safety Evaluation Report (SER), Reference 5, with some limitations and conditions that are considered in the MMR design. The fission product retention capability of TRISO fuel particles contributes to a "functional containment" whereby the TRISO particles serve as the first containment barrier when operated within the range of qualification parameters.
- The TRISO particles in MMR fuel are encased in an FCM pellet of SiC that provides an additional layer of retention for fission products, thereby achieving functional containment.
- The low power density of the active fuel region leads to slow fuel heat-up during loss of heat removal events.
- Low thermal power results in a small inventory available for release of the most limiting short-lived fission products for public safety, such as ^{131}I and ^{85}Kr. The increased inventory of long-lived fission products associated with a long core life is addressed by the defense-in-depth (DID) approach to functional containment.
- The low power rating also reduces the decay heat that must be removed in postulated accidents, simplifying passive decay heat removal.
- Heat transfer fluid used for core cooling during normal operation is an inert, chemically stable, single-phase gas (helium).
- Safety-related core cooling is passive and capable of maintaining fuel and component temperatures below limits with no helium, electrical power, or operator action.
- Intermediate heat transfer is performed by a molten salt loop that effectively isolates the reactor from transients in the adjacent plant power conversion system.
- The reactor is below ground. Although it does not have nor need a leak-tested containment building, it is surrounded by a concrete structure (the citadel) that provides DID for retention of fission products to the environment and provides protection against external hazards.

Table 1-1 provides a high-level summary of key design features of the MMR and, in comparison, to current LWRs.

Table 1-1. MMR Key Features and Differences from Operating LWRs

Feature	MMR	LWR	Remarks
Operating power level	Maximum research reactor allowed power (10 MW) for UIUC deployment	3000 to 4400 MW(t) (AP1000 3415 MW(t))	Full MMR power is less than decay heat of large LWR more than 24 hours after shutdown; short-lived fission product inventory is small
Heat transfer fluid	Helium – inert gas; single phase under all conditions; low stored energy	Water – also serves as moderator; scrubs fission products; high stored energy; undergoes phase change that causes high pressure and temperature in surrounding structure	Water coolant causes corrosion, and blowdown can damage safety systems by impingement, pressure, moisture, and temperature
Containment	Functional: TRISO integrity at high temperatures, with supplemental passive barriers for defense-in-depth, continuously confirmed by radiological monitoring while operating	Large containment building: subject to high pressure and temperature; many penetrations requiring active isolation, periodic leak testing, and maintenance	Functional containment is a set of barriers that effectively limit physical transport of radioactive material to the environment and serve as basis for the revised Principal Design Criteria (PDC) in RG 1.232 (Reference 6)
Safety-related ac power systems	None	Class 1E ac distribution and emergency diesel generators	MMR safety is provided by passive systems
Fuel form	Uranium oxycarbide (UCO) TRISO particles encased in FCM pellets in hexagonal graphite fuel blocks	Uranium dioxide pellets encased in zirconium alloy tubes	Insubstantial fission product release from MMR during operation or accidents
Fuel (^{235}U) enrichment	HALEU ≤ 19.75% ^{235}U	LEU < 5% ^{235}U	Both are low-enriched uranium; MMR higher enrichment provides for longer core life
Fuel damage temperature	> 3272˚F (1800˚C)	2200˚F (1204˚C)	Zirconium-water reactions start at about 1800˚F in LWRs
Emergency replenishment of coolant	None; fuel limits met for unmitigated primary system blowdown	ECCS needed	Must quickly recover LWR fuel with water if loss of coolant occurs
Hydrogen management	External/internal flooding might release hydrogen (graphite-water reaction)	Zirconium-water reaction produces hydrogen if clad exceeds 2200˚F	Acceptance criteria limit mass of LWR fuel clad reacted
Primary system corrosion mechanisms	While helium itself is non-corrosive, contaminants must be controlled to low levels to avoid degradation of graphite and other materials	Various types of stress corrosion cracking; boric acid corrosion (PWRs)	Helium is inert whereas hot water is corrosive unless water chemistry is carefully controlled

1.4. SCOPE

UUIC will be applying for a 10 CFR 50 CP and subsequent Class 104 Operating License (OL), using NUREG-1537 (Reference 1). This TR is part of pre-submittal activities to support NRC review of the Construction Permit Application (CPA) and Operating License Application (OLA).

UIUC is the license applicant and owner/operator of the MMR non-power utilization facility, with USNC as reactor designer/vendor/original equipment manufacturer and fuel supplier.

This TR considers regulations and guidance for preparing CPA and OLA for a research reactor facility in accordance with 10 CFR 50 (Reference 2) and NUREG-1537. The facility will be licensed under the provisions of 10 CFR 50.21(c) as a Class 104, non-power utilization facility. The MMR is also a non-light water reactor (non-LWR). These characterizations limit applicability of some NRC regulations. Reference 4 discusses the applicable NRC regulations for the MMR at UIUC.

This TR discusses the methodology to identify credible event sequences. A future TR will discuss the deterministic methodology used to (1) identify a Maximum Hypothetical Accident (MHA) that bounds the dose consequence associated the identified credible event sequence, (2) calculate the dose consequence associated with the MHA, and (3) analyze the limiting credible event sequence to demonstrate that the MHA dose consequence is bounding. This methodology will be consistent with NUREG-1537.

This TR also discusses the methodology for safety classification of SSCs. A future TR will provide the principal design criteria (PDC) for the UIUC MMR which establish necessary design, fabrication, construction, testing, and performance requirements for safety-related SSCs .

DID will be considered throughout the design process for the MMR and will be discussed in detail in the CPA and the OLA for the MMR.

1.5. NRC ACTION REQUESTED

UIUC is requesting NRC review and approval of the event sequence methodology in Section 2.0 and the safety classification methodology in Section 3.0 used for the MMR at UIUC.

Per the NRC draft guidance provided in Reference 3, a preliminary list of PIEs and SSC safety classifications are provided in Appendix A and B, respectively. These appendices, and the pipe break classification example in Section 3.0 are provided for information purposes to assist this TR review and not requested for approval at this time.

2.0 IDENTIFICATION OF EVENT SEQUENCES

2.1. REGULATORY FOUNDATION FOR EVENT SEQUENCE IDENTIFICATION METHODOLOGY

This section provides a summary of the applicable NRC regulatory requirements regarding the identification of event sequences for the MMR.

NRC reactor regulations mandate that safety analysis to assess the adequacy of the design during anticipated transient conditions must be performed and provided to the NRC. Specifically, 10 CFR 50.34(a)(4) states that a CPA under Part 50 must include:

> *"A preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety* ***during normal operations and transient conditions anticipated during the life of the facility****, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents."*
> [emphasis added]

Safety analysis of the event sequences identified using the methodology described in this section of the TR is intended to meet the requirements outlined in 10 CFR 50.34(a)(4). As discussed in Section 1.3, the safety analysis methodology used for the UIUC MMR deployment will be provided in a future TR.

2.2. DEFINITIONS RELATED TO EVENT SEQUENCE IDENTIFICATION

2.2.1. Postulated Initiating Event

A PIE is defined in UIUC MMR licensing basis as:

> A postulated event identified in design as capable of leading to anticipated operational occurrences or accident conditions.
>
> Note: A postulated initiating event is not an entire sequence itself; it is the event that initiates a sequence.

PIE types include:

- piping system breaches,
- transients (i.e., non-pipe breach reactor events such as reactivity additions),
- internal hazard induced PIEs (e.g., reactor building fire), and
- external hazard induced PIEs (e.g., severe weather).

2.2.2. Event Sequence

An event sequence is defined as:

1) a PIE or combination of PIEs that initially perturbs the plant,

2) the resulting response following the PIE(s), and

3) the resulting well-defined end state.

For the deterministic licensing approach used for the UIUC MMR deployment, only SSCs classified as safety-related using the methodology in Section 3.0 are credited to respond to PIEs. Additionally, the worst-case failure of any active component is assumed (i.e., single-failure criterion) when defining event sequences.

2.3. EVENT SEQUENCE IDENTIFICATION METHODOLOGY

2.3.1. Identify PIEs

The PIE identification methodology is split into three phases: (1) Initial phase, (2) Top-down phase, and (3) Bottom-up phase based on the maturity of the UIUC MMR design. The PIE list provided in Appendix A is based on the Initial phase methodology.

Initial Phase

An initial list of PIEs is first developed to provide conceptual information to the design process. This initial PIEs list is developed using historical information, regulatory documents, and engineering judgement.

The following historical and regulatory sources were considered and assessed against the MMR design to identify applicable PIEs, and screen out non-applicable PIEs where the technology was fundamentally different from that of the MMR, or not applicable:

- Fort St. Vrain Safety Analysis Report (Reference 8),
- MHTGR Licensing Basis Events (Reference 9), and
- CNSC, IAEA, and US NRC publications (References 10, 11, and 12).

Additional PIEs were also proposed, considered, and screened via engineering judgment.

For internal and external hazards, generic industry accepted hazard lists are used as a starting point for hazard identification.

The initial list of PIEs will be reconciled with the PIEs list using the Top-down and Bottom-up phase methodologies. Initial PIEs that are not captured by these two methodologies will be retained to ensure industry experience is considered.

Top-down Phase

In the Top-down phase of PIE identification, the design is mature enough to follow a "top-down" approach in determining PIEs. The following three fundamental safety functions have been identified for the MMR:

- Control of reactivity,
- Removal of heat from the reactor, and
- Control release of radioactive material that could exceed public dose limits.

By analyzing challenges to these functions, the PIEs identified at the initial phase can be confirmed or extended. Given the increased design maturity, specific initiating events can be identified. This process starts with the effect and determines a cause. For example, a reactivity excursion is the "effect", and improperly controlling reactivity is the possible cause that challenges the control of reactivity safety function. By cascading down the function tree, initiating events can be identified at the point where the function has been allocated to a human, automation system or hardware. Master logic diagrams (MLDs) will be used to perform the top-down phase methodology. Detailed description of the MLD methodology and results will be provided in the OLA.

Bottom-up Phase

In this phase it is possible to follow a "bottom-up" approach where the design is now defined in sufficient detail such that failures can more easily be determined. This phase acts as a verification and extension of the "top-down" approach. The bottom-up phase includes a unique methodology for each of PIE types listed in 2.2.1.

Pipe Breaches

Design descriptions are assessed to determine whether the breach of a piping system can result in anticipated operational occurrences or accident conditions. Generally, piping systems that contain reactor coolant and/or radioactive material, or breaches that could give rise to an internal hazard are retained for further analysis.

Specific pipe break locations are identified for retained piping systems. For helium pressure boundary (HPB) and molten salt system (MSS) breaches, all breach locations are retained. For other systems, breach locations can be qualitatively screened out depending on sources of radioactivity and hazard consequences/effect on plant.

Transients

Failure modes and effects analysis (FMEAs) will be performed to identify PIEs for transients (non-pipe breach reactor events). Detailed description of the FMEA methodology and results will be provided in the OLA.

Internal Hazards

An Internal Hazard Assessment identifies internal hazards associated with the standard MMR product, and project specific configuration aspects. A Hazard Analysis is performed, to identify hazard induced PIEs. Detailed description of the Internal Hazard Assessment and Analysis methodology and results will be provided in the OLA.

External Hazards

An External Hazard Assessment identifies external hazards associated with the UIUC site. A Hazard Analysis is performed, to identify hazard induced PIEs. Detailed description of the External Hazard Assessment and Analysis methodology and results will be provided in the OLA.

2.3.2. Screen PIEs

In alignment with NUREG-1537 (Reference 1), only credible events, on a deterministic basis, are considered (an MHA will be identified that bounds the consequences of all credible events). Therefore, PIEs that are determined to be non-credible are screened out.

For PIEs related to an SSC failure (e.g., pipe breach), the engineering design rules for SSCs associated with each identified PIE will be reviewed. A determination will be made if the PIE is credible or not based on the specifics of the SSC failure and/or the engineering design rules applied to the SSC. For example, a double-ended guillotine break at the hot gas duct PIE would be screened out as non-credible because it’s expected the hot gas duct is designed to American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III (Reference 7), which practically eliminates large ruptures from occurring.

For external hazards, a determination of PIE credibility will be made based on the specifics of the external hazard and/or available hazard frequency information from the External Hazard Assessment.

If sufficient information does not exist to screen out a PIE, the PIE is retained as part of the plant licensing basis until it can be properly assessed.

2.3.3. Defining Event Sequences

After PIEs are identified, the safety classification methodology in Section 3.0 is used to determine which SSCs are required to mitigate the consequences of the PIEs. SSCs that are classified as safety-related determine the plant response to each PIE (or combination of PIEs) to define event sequences deterministically.

Additionally, the subsequent worst-case single failure of an active component will be determined for each event sequence and assumed when performing safety analysis.

2.3.4. Grouping of Event Sequences

Once credible event sequences are identified, they are grouped using the accident categories provided in NUREG-1537 (Reference 1). Given that the accident categories were created for LWR technologies, Table 2-1 provides the corresponding event sequence category for the MMR. Event sequence categories may be added and/or removed as the event sequence list matures.

Table 2-1. Grouping of MMR Event Sequences

NUREG-1537 Accident Categories	MMR Event Sequence Categories
MHA	MHA
Insertion of Excess Reactivity	Insertion of Excess Reactivity
Loss of Coolant	D-LOFC (depressurized loss of forced cooling) with air ingress
Loss of Coolant Flow	P-LOFC (pressurized loss of forced cooling)
Mishandling or Malfunction of Fuel	Mishandling or Malfunction of Fuel
Experiment Malfunction	N/A (No in-core experiments)
Loss of Normal Electrical Power	Loss of Normal Electrical Power
External Events	Internal and External Hazards
Mishandling or Malfunction of Equipment	Mishandling or Malfunction of Equipment

3.0 CLASSIFICATION OF SYSTEMS, STRUCTURES, AND COMPONENTS

3.1. REGULATORY FOUNDATION FOR CLASSIFICATION OF SYSTEMS, STRUCTURES, AND COMPONENTS

This section provides a summary of the applicable NRC regulatory requirements regarding the safety classification of SSCs for the MMR plant.

10 CFR 50.2 defines safety-related SSCs as those SSCs that are:

> *"…relied upon to remain functional during and following design basis events to assure:*
>
> 1) *The integrity of the reactor coolant pressure boundary;*
> 2) *The capability to shut down the reactor and maintain it in a safe shutdown condition; or*
> 3) *The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.…"*

The definition of safety-related SSCs in 10 CFR 50.2 is applicable to water-cooled reactors. Per Reference 4, Attachment 1, the UIUC MMR will "meet the intent" of the definition of safety-related SSCs. Alternative safety functions appropriate for the UIUC MMR are provided in Section 3.2.1 to meet the intent of 10 CFR 50.2.

As discussed in Section 2.1, 10 CFR 50.34(a)(4) states that a CPA under Part 50 must include:

> *"A preliminary analysis and evaluation of the design and **performance of structures, systems, and components** of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and **the adequacy of structures, systems, and components provided for the prevention** of accidents and the mitigation of the consequences of accidents."* [emphasis added]

There are corresponding requirements regarding the OLA.

3.2. DEFINITIONS RELATED TO SSC CLASSIFICATIONS

3.2.1. Safety Functions

A safety function is a specific purpose that must be accomplished for safety for a facility or activity to prevent or to mitigate radiological consequences of normal operation, anticipated operational occurrences, and accident conditions.

The UIUC MMR safety classification methodology considers the fundamental safety functions below in lieu of the functions for safety-related SSCs in 10 CFR 50.2.

- Control of reactivity,
- Removal of heat from the reactor, and
- Control release of radioactive material that could exceed public dose limits.

3.2.2. Safety Classification Groups

The following two classification groups are used for the UIUC MMR safety classification methodology:

- **Safety-Related (SR):** SSCs that have an impact on safety and are relied upon to remain functional to meet the three safety functions in Section 3.2.1 during and following all event sequences part of the plant design-basis.
- **Non-Safety-Related (NSR):** SSCs not required to remain functional to meet the three-safety function in Section 3.2.1.

3.3. METHODOLOGY FOR SSC SAFETY CLASSIFICATION

This section defines the approach used for determining the safety classification and applicability of design requirements for SSCs. The purpose of safety classification is to ensure that SSCs are designed, fabricated, inspected, tested, operated, and maintained based on their roles in preventing and/or mitigating event sequences. This process is iterative during the design. The preliminary SSC safety classifications are provided in Appendix B.

3.3.1. Identify Limiting PIEs Relevant to Safety Classification

The limiting PIEs will be identified and serve as an input in the safety classification methodology.

3.3.2. Identify SSCs Required to Achieve Safety Functions During Event Sequences

The next step in the classification process is to identify the SSCs that are required to achieve each of the safety functions for the limiting PIEs. This analysis is performed by separately evaluating each limiting PIE. Each analysis considers all three safety functions: 1) control of reactivity, 2) removal of heat from the reactor, and 3) control release of radioactive material that could exceed public dose limits.

Pipe Breach Example

Table 3-1 identifies the systems that contain SSCs necessary to perform the safety functions for a hypothetical pipe breach of the HPB. A narrative of the pipe breach analysis, discussing how determinations of safety-related SSCs, is provided in the following table. Note that the SSC names and classifications provided in this section are preliminary and provided as an example to demonstrate the safety classification methodology. The SSC names and classifications are subject to change as the design matures.

Table 3-1. Safety Function Analysis for Pipe Breach PIE

	Systems																									
	JK – Reactor Core	**JR – RPS**	**JA – Reactor Vessel**	**JB – Core Support Structure**	**JD – RCSS**	**KA – RCCS**	**UJ – Citadel**	**B0 – NP Elect. Aux. Power**	**JT – SPDS**	**JY – Monitoring**	**XS – Access Control**	**Y0 – Comm. & Info. Systems**	**C0 – NP Control, Data, & Instr.**	**JG – Molten Salt**	**JS – RCS**	**JE – HTS**	**KB – He Purification Supply**	**KL – HVAC**	**KM/N/P – Waste Treatment**	**KT – Drains**	**KU – Sampling**	**UK – Nuclear Building**	**VA – NP Storage**	**XF – Earthing**	**XG – Fire System**	**XK – Chilled Water**
Provides Safety Function?	Y	Y	Y	Y	Y	Y	Y	N	N	N	N	N	N	N	N	N	N	N	N	N	N	N	N	N	N	N

Safety Function Analysis:

Y = Yes; System contains an SSC that is credited to provide at least one safety function; therefore, is safety-related

N = No; System does not contain an SSC that is credited to provide at least one safety function

Control of Reactivity:

The Reactor Protection System (RPS) and Reactivity Control and Shutdown System (RCSS) are required to be safety-related to trip the reactor after the break and maintain a subcritical state post reactor-trip.

The Reactor Core is required to be safety-related because the fuel and graphite structures provide core geometry to allow insertion of the control rods of the reactivity control and shutdown system to insert under gravity. To ensure core geometry it is essential that the core is retained in position, hence the Core Support Structure, Vessel and Citadel provide the structural support for the core to ensure core geometry and rod insertion.

Removal of Heat from Reactor:

When a pipe breach occurs, forced cooling from the helium is quickly lost. Only passive SSCs are credited to provide the safety function to remove decay heat from the reactor for this PIE.

Passive components of the Reactor Cavity and Cooling System (RCCS), i.e., water in the RCCS standpipes, are required to be safety related to provide heat capacity to remove the heat following the reactor trip.

The Reactor Core, Core Support Structure, Reactor Vessel, and Citadel are all required to be safety-related as these SSCs' geometry and heat capacity are credited to reject heat radially from the fuel to the surrounding soil and bedrock in the longer term to remove decay heat.

Control release of radioactive material that could exceed public dose limits:

The Reactor Core is required to be safety-related to meet this safety function. Specifically, the barriers to fission product release (i.e., retention layers in TRISO particles and FCM) are credited to confine radioactive material.

All other systems are non-safety related.

3.3.3. Assign SSCs to Classification Groups

After the safety function analysis is completed for the limiting PIEs, any SSC that is required to provide a safety function for a PIE is classified as safety-related. All other SSCs are classified as non-safety-related.

3.3.4. Apply Engineering Design Rules

The PDC, which will be provided in a future TR, will establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components. Details of engineering design rules applied to safety-related SSCs to satisfy the PDC will be provided in the CPA.

4.0 REFERENCES

1. Nuclear Regulatory Commission, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors Standard Review Plan and Acceptance Criteria," NUREG-1537, Part 1 and Part 2, February 1996.
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A PRELIMINARY LIST OF PIES

A preliminary list of postulated initiating events and related assumptions for the UIUC MMR licensing basis are provided in this appendix. This list was generated by using the Initial Phase methodology described in Section 2.3.1. The final list of PIEs will be provided in the OLA.

A.1 INSERTION OF EXCESS REACTIVITY

- Molten Salt Maximum Flow (Overcooling Event)
  - Molten salt flow instantly changes from normal parameters to minimum inlet temperature and maximum flow rate simultaneously.
- Spurious withdrawal of Single Control Rod
  - Single, maximum worth, control rod is spuriously withdrawn from current position at limiting withdrawal speed.
- Spurious withdrawal of Control Rod Group
  - Three control rods are spuriously withdrawn from their current position at limiting withdrawal speed.
- Simultaneous withdrawal of All Actuatable Rods
  - All control rods are spuriously withdrawn at the limiting withdrawal speed.
- Single Rod Ejection
  - A single, maximum worth control rod is ejected from its current position instantaneously due to a failure in the pressure boundary and a failure to restrain the control rod.

A.2 D-LOFC

Note that the impact of air ingress into the primary coolant loop must be evaluated for D-LOFC event sequences.

- Un-isolated small breach in HPB
- Un-isolated small pipe breach outside Citadel
- Small Intermediate Heat Exchanger (IHX) Leak
    - A small IHX leak occurs when pressurized helium leaks into the salt side.
- Breach in largest connecting pipe to pressure boundary

A.3 P-LOFC

- Blockage of Helium Coolant Channel
  - Blockage in helium flow in a single fuel element
- Loss of Direct Reactor Cooling (Loss of Primary Helium Flow)
- Molten Salt Flow Stop
- Loss of Secondary Heat Sink
- Total Loss of Forced Flow (Total Loss of Primary and Secondary Flow)

A.4 MISHANDLING OR MALFUNCTION OF FUEL

- Inadvertent Load and Operation of a Fuel Element in an Improper Position

A.5 INTERNAL AND EXTERNAL HAZARDS

Internal and external hazards considered during the design of the UIUC MMR are provided in this section.

Generic Internal Hazards

- Internal Fires
- Internal Explosions
- Internal Missiles and Pipe Breaches
- Internal Flooding
- Heavy Load Drop
- Electromagnetic Interference
- Release of Hazardous Substances inside the Plant

Generic External Hazards

- External Floods
- Hazardous Substances
- Aircraft Crash
- EMF including Solar Storms
- Biological Phenomena
- Meteorological disturbances (such as hurricane, tornado, or flood)
- Seismic event
- Mechanical impact or collision with building
- Event caused by humans, such as explosion or toxic release near the reactor building

B PRELIMINARY SSC SAFETY CLASSIFICATIONS

The preliminary systems and structures safety classifications are provided in Table B-1. The methodology discussed in Section 3.3 is used to generate the classifications. Sub-system safety classifications are provided in Table B-2.

Table B-1. Preliminary Systems and Structures Safety Classifications

System/ Structure Code	System/Structure	UIUC MMR Safety Classification
JK	Reactor Core	Safety-Related
JR	Reactor Protection System (RPS)	Safety-Related
JA	Reactor Vessel	Safety-Related
JB	Core Support Structure	Safety-Related
JD	Reactivity Control and Shutdown System (RCSS)	Safety-Related
KA	Reactor Cavity Cooling System (RCCS – Only Passive Cooling Components)	Safety-Related
UJ	Citadel	Safety-Related
B0	Nuclear Plant (NP) Electrical Auxiliary Power	Non-safety-related
C0	NP Controls, Data, and Instrumentation	Non-safety-related
JE	Heat Transport System (HTS)	Non-safety-related
JG	Molten Salt (MS) System	Non-safety-related
JS	Reactivity Control System (RCS)	Non-safety-related
JT	Safety Parameter Display System (SPDS)	Non-safety-related
JY	Radiation Monitoring System	Non-safety-related
KB	Helium Purification and Supply System	Non-safety-related
KL	HVAC System	Non-safety-related
KM/N/P	Waste Treatment	Non-safety-related
KT	Drains	Non-safety-related
XF	Earthing	Non-safety-related
XG	Fire System	Non-safety-related
XS	Access Control	Non-safety-related
Y0	Communication and Information Systems	Non-safety-related
KU	Sampling	Non-safety-related
UK	Nuclear Building	Non-safety-related
VA	NP Storage	Non-safety-related
XK	Chilled Water	Non-safety-related

Table B-2. Sub-system Safety Categorization

System/ Structure Code	Sub Systems	Safety Class
B0	Electrical – Normal	Non-Safety-Related
	Electrical – Essential loads	Non-Safety-Related
JA	Reactor Vessel	Safety-Related
	Cross duct	Non-Safety-Related
	Intermediate Heat Exchanger (IHX) Vessel	Non-Safety-Related
	Pressure Relief Valve (PRV)	Non-Safety-Related
	Reactor Vessel Support	Safety-Related
JD	Controls Rods	Safety-Related
	Control Rod Drive Mechanisms (CRDM)	Safety-Related
	Controller	Non-Safety-Related
JE	Helium Circulator	Non-Safety-Related
	IHX	Non-Safety-Related
	Hot Gas Duct (HGD)	Non-Safety-Related
JG & KO	MS Piping	Non-Safety-Related
	MS Valves	Non-Safety-Related
	MS Drain Tank	Non-Safety-Related
	MS Cold Pumps	Non-Safety-Related
	MS Hot/Cold Tanks	Non-Safety-Related
JR	Trip Breaker	Safety-Related
	Flux monitor	Safety-Related
	Pressure Monitor	Safety-Related
	Temp Monitor	Safety-Related
	He Flow Monitor	Non-Safety-Related
	Inverter supply	Safety-Related
	Control Cabinet	Safety-Related
	Readout Monitor	Non-Safety-Related
JY	Helium Activity Monitor	Non-Safety-Related
	Salt Activity Monitor	Non-Safety-Related
	Radiation Monitor	Non-Safety-Related