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

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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.

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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 postulated initiating events (PIEs) and SSC 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 1.4~~1.3~~, the safety analysis methodology used for the UIUC MMR deployment will be provided in a future TR.

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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:

were created for LWR technologies, Table 2-1~~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

Event sequences will also be grouped under a limiting event sequence, such that the group of event sequences will rely on the same set of responding SSCs to perform safety functions.

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

Note, control of reactivity encompasses reactor shutdown.

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~~ Limiting Event Sequences Relevant to Safety Classification

The ~~limiting PIEs~~ limiting event sequences 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~~limiting event sequences. This analysis is performed by separately evaluating each ~~limiting PIE~~limiting event sequence. 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~~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

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

	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~~event sequence.

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~~limiting event sequences, 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.