Technology Inclusive Content of Application Project
For Non-Light Water Reactors

TerraPower Molten Chloride Reactor Experiment
TICAP Tabletop Exercise Report

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Abstract

Non-light water reactor (non-LWR) technologies will play a key role in meeting the world's future energy needs and will build on the foundation established by the current light water reactor (LWR) nuclear energy fleet. Given the long timeframe and significant financial investment required to mature, deploy, and optimize these technologies, an efficient and cost-effective non-LWR-licensing framework that facilitates safe and cost-effective construction and operation is a critical element for incentivizing private sector investment. The Technology Inclusive Content of Application Project (TICAP) is an important step in establishing that licensing framework. This Department of Energy (DOE) cost-shared, owner/operator-led initiative will produce guidance for developing content for specific portions of the Nuclear Regulatory Commission (NRC) license application Safety Analysis Report (SAR) for non-LWR designs.

The portions of the SAR on which this work will focus are those addressed in the Nuclear Energy Institute (NEI) publication NEI 18-04, "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development." The TICAP guidance will help ensure completeness of information submitted to NRC while avoiding unnecessary burden on the applicant and rightsizing the content of application commensurate with the complexity of the design being reviewed.

TICAP will generate a number of products culminating in an NRC-endorsable NEI document providing guidance for key elements of the content of an advanced reactor license application. This report describes the tabletop exercise conducted with TerraPower to explore the application of the draft TICAP guidance to the safety case for the Molten Chloride Reactor Experiment (MCRE) design. Example content for parts of SAR Chapter 3 (Licensing Basis Events), Chapter 5 (Safety Functions, Design Criteria, and SSC Classification), and Chapter 6 (Safety-Related SSC Criteria and Capabilities) were developed, and feedback from the development of this content informed revisions to the TICAP guidance for the purpose of optimizing the guidance document. In addition to the example SAR content, this report provides additional context about the MCRE design and safety case and documents the major lessons learned about the TICAP guidance during this tabletop exercise.

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List of Abbreviations

AOO	Anticipated Operational Occurrence
ARDC	Advanced Reactor Design Criteria
BDBE	Beyond Design Basis Event
CDC	Complementary Design Criteria
DBE	Design Basis Event
DID	Defense-in-Depth
DOE	Department of Energy
F-C	Frequency-Consequence
FSF	Fundamental Safety Function
GDC	General Design Criteria
HRS	Heat Rejection System
INL	Idaho National Laboratory
LBE	Licensing Basis Event
LMP	Licensing Modernization Project
LOOP	Loss of Offsite Power
LWR	light water reactor
MCFR	Molten Chloride Fast Reactor
MCRE	Molten Chloride Reactor Experiment
MHTGR	General Atomics Modular High Temperature Gas-Cooled Reactor
MSR	Molten Salt Reactor
NEI	Nuclear Energy Institute
non-LWR	non-light water reactor
NRC	Nuclear Regulatory Commission
NSRST	Non-Safety-Related with Special Treatment
PCS	Primary Coolant System
PCS-HX	Primary Cooling System Heat Exchanger
PDC	Principal Design Criteria
PRA	Probabilistic Risk Assessment
RCS	Reactor Core System
RCS-P	Reactor Core System Pump
RFDC	Required Functional Design Criteria
RIPB	risk-informed and performance-based
RPS	Reactor Protection System
RSF	Required Safety Function
RXE	Reactor Enclosure System
RXE-HX	Reactor Enclosure System Heat Exchanger
SAR	Safety Analysis Report
SR	Safety-Related
SRDC	Safety-Related Design Criteria
SSCs	Structures, Systems, and Components
TICAP	Technology Inclusive Content of Application Project
ZTB	ZPPR Test Bed

1.0 INTRODUCTION AND BACKGROUND

1.1 TICAP Description

Non-light water reactor (non-LWR) technologies will play a key role in meeting the world's future energy needs and will build on the foundation established by the current light water reactor (LWR) nuclear energy fleet. Given the long timeframe and significant financial investment required to mature, deploy, and optimize these technologies, an efficient and cost-effective non-LWR-licensing framework that facilitates safe and cost-effective construction and operation is a critical element for incentivizing private sector investment. The Technology Inclusive Content of Application Project (TICAP) is an important step in establishing that licensing framework. This Department of Energy (DOE) cost-shared, owner/operator-led initiative will produce guidance for developing content for specific portions of the Nuclear Regulatory Commission (NRC) license application Safety Analysis Report (SAR) for non-LWR designs.

The portions of the SAR on which this work will focus are those addressed in the Nuclear Energy Institute (NEI) publication NEI 18-04, "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development."[1] The TICAP guidance will help ensure completeness of information submitted to NRC while avoiding unnecessary burden on the applicant and rightsizing the content of application commensurate with the complexity of the design being reviewed.

Existing LWRs are the country's largest source of emissions-free, dispatchable electricity, and they are expected to remain the backbone of nuclear energy generation for years to come. However, as the energy and environmental landscape has evolved, interest has grown in advanced nuclear energy systems that promise superior economics, improved efficiency, greater fissile-fuel utilization, reduced high-level waste generation, and increased margins of safety. In addition to electricity generation, these technologies can expand the traditional use of nuclear energy by providing a viable alternative to fossil fuels for industrial process heat production and other applications.

The current regulatory framework for nuclear reactors was developed over decades for LWRs using zirconium-clad uranium oxide fuel and coupled with the Rankine power cycle. Many advanced, non-LWRs are in development, with each reactor design differing greatly from the current generation of LWRs. For example, advanced reactors might employ liquid metal, gas, or molten salt as a coolant, enabling them to operate at lower pressures but higher temperatures than LWRs. Some employ a fast rather than a thermal neutron spectrum. A range of fuel types is under consideration, including fuel dissolved in molten salt and circulated throughout the primary coolant system. In general, advanced reactors emphasize passive safety features that do not require rapid action from powered systems to prevent radionuclide releases. Given these major technical differences, changes to the current regulatory framework are needed for the deployment of advanced reactor designs.

Therefore, DOE authorized TICAP, a utility-led initiative to improve the effectiveness and efficiency of NRC's current regulatory framework. The initiative recognizes that significant levels of industry input and advocacy are needed in collaboration with NRC to enable the regulatory changes needed for advanced reactors.

The goal of TICAP is to develop license application content guidance with the following attributes:

- Technology inclusive to be generically applicable to all non-LWR designs
- Risk-informed and performance-based (RIPB) to:
  - Ensure the NRC review is focused on information that impacts the safety case of reactors.
  - Create coherency and consistency in the scope and level of detail requirements in the license application for various advanced technologies and designs.
  - Provide for flexibility during construction.
  - Encourage innovation by focusing on the final results as opposed to the pathway taken to achieve the results.

This modernized, technology inclusive RIPB license application content will advance:

- The NRC’s longstanding focus on and commitment to continuous improvement.
- The industry (developers and owners/operators) goal of having a safety-focused review that minimizes the burden of generating and supplying safety-insignificant information.
- The NRC and industry objective of reaching agreement on how to implement reasonable assurance of adequate protection for non-LWRs.
- NRC’s stated objective and policy statement regarding the use of risk-informed decision-making to remove unnecessary regulatory burden.

TICAP will build on the success of the Licensing Modernization Project (LMP) that produced NEI 18-04. That document presented a modern, technology inclusive RIPB process for selection of Licensing Basis Events (LBEs); safety classification of Structures, Systems, and Components (SSCs) and associated risk-informed special treatments; and determination of Defense-in-Depth (DID) adequacy for non-LWRs. The TICAP application guidance will focus on the portion of the application related to LMP and the applicant’s safety case. Ultimately, the information presented in the application must demonstrate reasonable assurance of adequate protection of public health and safety.

1.2 Purpose of TICAP Tabletop Exercises

TICAP will generate a number of products culminating in an NRC-endorsable NEI document providing guidance for key elements of the content of an advanced reactor license application. Figure 1 provides a list of the products with the subject of this report highlighted. Each of these products is described below.

Figure 1. TICAP Products

- Fundamental Safety Functions (FSFs) Definition—A set of high-level functions, labeled as Fundamental Safety Functions (also known as performance objectives), will be defined that, when accomplished, satisfy the public safety objective of the regulation. The FSFs are applicable, as relevant, throughout the lifetime of the facility for which the license is being submitted.
- Regulation Mapping to Fundamental Safety Functions—The underlying safety basis of the current regulatory requirements will be identified and will be mapped to the FSFs.
- SAR Options Assessment—The current SAR content will be reviewed to identify those sections that will be the subject of rightsizing in this project. It is important to note that only those sections/elements that are part of both the LMP's processes and their expected outputs will be targets of this project.
- LMP-Related Safety Case—The input (e.g., data, design information, analytical programs, and tools such as a probabilistic risk assessment) used to generate and select the LBEs, classify SSCs, and determine DID adequacy, as well as the outputs (i.e., the required safety functions, SSC classification, required functional design criteria), will be delineated.
- Differences Between Licensing Paths—It is recognized that different applicants may select different licensing paths (e.g., combined construction and operating license, construction permit/operating license, or design certification) to deploy their reactor designs. To facilitate the execution of these options, the scope, level of details, and the maturity of the information that needs to be provided for several typical licensing paths will be defined.
- *Tabletop Exercises (including this document)—To improve the efficacy of the proposed process, some elements of the recommendations will be subjected to trial use tests. This effort will be supplemented by discussions with user communities (e.g., developers and/or prospective site applicants) in order to obtain the maximum independent insights on the proposed processes.*
- Formulation of Technology Inclusive Content of Application—The formulation of and the basis for developing application content will be based on previous products, FSFs Definition, Regulation Mapping to FSFs, SAR Options Assessment, and the LMP-Related Safety Case.
- NEI Content of Application Guidance Document—The results of the above deliverables/activities will be finalized in an endorsable NEI document. This deliverable will be an integrated product of various predecessor products that have been adjusted for the purposes of the Guidance Document.

Each of the four tabletop exercises explored the application of a unique subset of the draft TICAP guidance to a different non-LWR design. These exercises resulted in four separate tabletop reports that document example SAR content developed using the draft TICAP guidance, additional context about the specific design and safety case necessary to understand the example SAR content, and the major lessons learned for a given exercise.

This report presents the design and safety case details, example SAR content, and lessons learned for the tabletop exercise conducted on the Molten Chloride Reactor Experiment (MCRE) design coordination with TerraPower. The MCRE tabletop exercise explored the development of example SAR content for parts of Chapter 3 (Licensing Basis Events), Chapter 5 (Safety Functions, Design Criteria, and SSC Classification), and Chapter 6 (Safety-Related SSC Criteria and Capabilities), using the draft TICAP guidance.

1.3 Linkage to Other LMP and TICAP Documents

Table 1 displays relevant products from the LMP and TICAP efforts and describes the relationship to the MCRE tabletop exercise and this report.

Table 1. Relationship of Relevant LMP and TICAP Documents to MCRE Tabletop Exercise

Document	Relationship to MCRE Tabletop Exercise
NEI 18-04[1], [2]	The LMP approach documented in NEI 18-04 (and endorsed by the NRC in Regulatory Guide 1.233) is the basis for the approach used by TerraPower to develop a RIPB safety case for the MCRE design. The structure of the TICAP guidance (and, as a result, the SAR content developed using the TICAP guidance) leverages the concepts and tasks outlined in NEI 18-04.
LMP Reports[3], [4], [5], [6]	Four topical reports expand upon the approach described in NEI 18-04 in the areas of (1) Probabilistic Risk Assessment (PRA) approach; (2) selection and evaluation of Licensing Basis Events; (3) safety classification and performance criteria for SSCs; and (4) RIPB evaluation of Defense-in-Depth. The more detailed discussions of these topics and the specific examples in the reports were useful to the Xe-100 tabletop exercise, including the identification of plant programs.
TICAP Guidance Document[7]	The TICAP Guidance Document was used to identify the content, structure, and level of detail of the example SAR content developed for the MCRE tabletop exercise. The draft version of the TICAP guidance used for the exercise was an early revision; however, this revision has since been superseded (including changes made as a result of the tabletop exercises) and is not yet publicly available.
Application of the Licensing Modernization Project Approach to the Authorization of the Versatile Test Reactor[8]	In reality, MCRE is a small test reactor that is seeking DOE authorization for construction and authorization rather than NRC licensing. For this tabletop, the MCRE safety case was presented using NRC terminology, consistent with NEI 18-04. Although this conference paper is not a product of the LMP initiative, the discussion of how the Versatile Test Reactor team has used the NEI 18-04 approach within the DOE authorization framework was useful to translate the MCRE safety case into language consistent with the NRC licensing framework.

1.4 MCRE Tabletop Exercise Objectives, Scope, and Deliverables

The major deliverables of the MCRE TICAP tabletop exercise are: (1) a TICAP tabletop exercise meeting observed by NRC staff resulting in feedback from the staff on the concepts explored during the exercise and (2) this report, which includes example SAR content, additional context to understand the examples, and lessons learned.

Within the broader TICAP effort, the tabletop exercises had the following high-level objectives:

1. Technically improve TICAP guidance by obtaining input from advanced reactor developers
2. Maximize the usefulness of the guidance by providing examples for future users
3. Improve stakeholder confidence with the Guidance Document for NRC endorsement

The first objective was largely achieved during regular working meetings from November 2020 to April 2021 between the TICAP team and the TerraPower tabletop team. During these working meetings, the focus was on discussing the example SAR content as it was developed and gathering feedback from the TerraPower team on how the draft TICAP guidance might be revised to improve its usability. The major feedback from these working meetings is also documented in this report.

The purpose of this report is mostly focused on the second objective of the MCRE tabletop exercise. Taking into consideration the scope of the other TICAP tabletop exercises, the scope of the MCRE exercise included exploration of LBE narratives (SAR Chapter 3) and discussion of how the Safety-Related (SR) SSCs fulfill the Required Safety Functions (RSFs) and Principal Design Criteria (PDC) (SAR Chapter 5). Discussion of how the Non-Safety-Related with Special Treatment (NSRST) SSCs fulfill the Complementary Design Criteria (CDC) (SAR Chapter 5) was also explored. Example SAR content that would belong in these chapters of the SAR is displayed in the appendices of this report. Additionally, some concepts that relate to a SAR developed using the TICAP guidance are discussed in the body of the report, including an exercise to compare the MCRE PDC identified using the RIPB approach to PDC identified by NRC guidance in Regulatory Guide 1.232. It should be noted that the format and structure are not fully representative of how this information would be presented in a SAR.

The third objective of the MCRE tabletop exercise was achieved by a working meeting between the TICAP team and the MCRE tabletop team on March 31, 2021, that was observed by NRC and Idaho National Laboratory (INL) staff. During the meeting, the example SAR content was discussed, including major feedback items that provided the observers an opportunity to explore the scope and level of detail for descriptions of these portions of an LMP-based affirmative safety case. The observations of the NRC and INL staff were made publicly available[*] following the meeting, in addition to the feedback of the TICAP team on these observations.

[*] https://www.nrc.gov/reactors/new-reactors/advanced/details.html#licensing

1.5 Report Organization

Section 1 of this report presents background information on the TICAP effort and context for how the MCRE tabletop exercise supports the broader objectives of the project. Section 2 provides context related to the MCRE design effort, such as the role of MCRE within the Molten Chloride Fast Reactor (MCFR) development plan and the maturity of the MCRE design and safety case at the time of the MCRE tabletop exercise. Section 3 contains technical information that supports a better understanding of the draft SAR content that is presented in the appendices. The technical content in the body of the report (i.e., Section 3 of this report) includes technical analyses that are outside of the scope of the exercise and assumptions that were made to produce the draft SAR content. The observations, experiences, and lessons learned from this tabletop exercise are presented in Section 4. Finally, Appendix A displays example content for SAR Chapter 3, Appendix B displays example content for SAR Chapter 5, and Appendix C displays example content for SAR Chapter 6—all developed in accordance with the draft TICAP guidance.

2.0 DEMONSTRATION OVERVIEW

2.1 Summary of Demonstration Activities

The MCRE TICAP tabletop exercise took place between November 2020 and April 2021. The tabletop team largely relied upon the MCRE conceptual design documentation that has been generated to date. A primary purpose of this exercise was to gain a thorough understanding of portions of the COA for NRC licensing of a commercial nuclear reactor using the TICAP guidance and to allow the process for the development of this content to be fully vetted and offer improvement of the guidance being developed. The MCRE design is currently envisioned as an experimental reactor located at a DOE facility. Because there are differences in terminology between the NRC and DOE regulatory frameworks, an effort was made to correlate these differences as appropriate. For example, the tabletop report uses NRC terminology (e.g., "safety-related") instead of DOE terminology (e.g., "safety class").

Additionally, the LMP uses three different frequency categories that are similar to those used for DOE licensing. Events are categorized based on frequencies between Anticipated Operational Occurrences (AOOs), Design Basis Events (DBEs), and Beyond Design Basis Events (BDBEs). These categories generally line up with DOE frequency categories: anticipated events, unlikely events, extremely unlikely events, and beyond extremely unlikely events. This general alignment is shown in Table 2. BDBEs are considered to have lower frequencies than Extremely Unlikely events. Similar to DOE guidance to not use 10^{-6} as a strict cutoff for consideration, those events less frequent than BDBEs are generally excluded from consideration in the Safety Basis Events list for licensing, but the results are retained and assessed to ensure there are no unacceptable cliff edge events below the frequencies typically considered. This will be discussed in greater detail within Section 3.3 of this report.

Table 2. LMP and DOE Event Category Comparison

LMP Process Event Category	Frequency	DOE Frequency Category	Frequency
AOO	$>10^{-2}$	Anticipated	$>10^{-2}$
DBE	10^{-2} to 10^{-4}	Unlikely	10^{-2} to 10^{-4}
BDBE	10^{-4} to 5×10^{-7}	Extremely Unlikely	10^{-4} to 10^{-6}
Not categorized but retained in PRA	$<5 \times 10^{-7}$	Beyond Extremely Unlikely	$<10^{-6}$

A tabletop meeting, including observation by NRC staff, was held virtually on March 31, 2021. The focus of the meeting was to finalize the example SAR content presented in the appendices of this report. The meeting allowed for direct feedback from the NRC on areas recommended for improvement. An additional intent of this meeting was to familiarize the NRC with the MCFR technology, the anticipated performance of the reactor and other systems, the ongoing activities to develop the MCFR technology, and the MCRE design and analysis processes.

2.2 MCRE Background

MCRE's mission is to reduce the technical, licensing, and capital risk of TerraPower's MCFR technology. Fundamentally, the MCRE is a physics experiment that is focused on collecting data for MCFR-specific phenomena, including (but not limited to) the following:

- Fuel salt handling, including synthesis, melting, transfer, and disposal
- Liquid fueled reactor approach to criticality, startup, fine reactivity control, and shutdown
- Reactor control and stability of a flowing fuel system with a very low effective delayed neutron fraction
- Inherent load-following capability

Beyond the primary objectives above, the MCRE will provide invaluable practical experience in the design and operation of fast-spectrum molten salt reactors. Specific challenges addressed by the MCRE include:

- Criticality (k-eff) uncertainty due to nuclear data and chloride-based fuel salt thermophysical properties
- Materials and components performance within concurrent high temperature and high radiation environment
- Fuel salt pump design, including bearings and seals
- Fission gas capture and processing design

MCRE is shown in the context of the MCFR technology development roadmap in Figure 2.

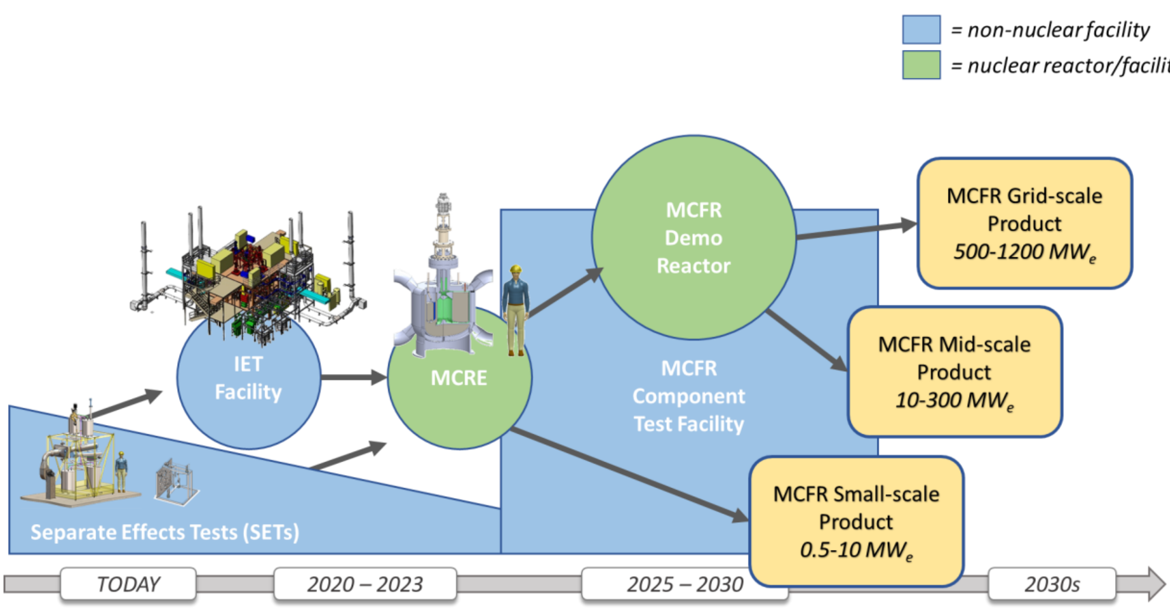

Figure 2. MCFR Commercialization Timeline

Presently, separate effects tests are ongoing, which include natural circulation microloops, a pumped isothermal coolant loop, and pumped polythermal coolant loop, as well as a larger pumped isothermal coolant loop and pumped polythermal coolant loop. A larger natural circulation loop, known as the milliloop, is also being designed. Beyond separate effects tests, the Integrated Effects Test, as part of the DOE ARC15 award, is in final design and is being built at TerraPower's Everett Lab. Eventually, MaSTiF, a molten salt component testing facility, will be designed, build, and operated. Beyond MaSTiF and MCRE, the next critical MCFR will be an MCFR Demonstration Reactor that is expected to start at 30 MWth with the intention to uprate its thermal power following the initial demonstration.

The MCFR program aims to reduce fast spectrum chloride-based fuel salt reactor technology development risks with the MCRE. The MCRE is expected to confirm that an MCFR reactor with the following unique characteristics is fundamentally stable, controllable, and safe:

- Flowing (pumped) molten fuel salt around a flow circuit within a pool
- Low effective delayed neutron fraction
- Low prompt neutron lifetime
- 600°C and higher operating temperature
- Molten fuel salt transfer between the reactor core and drain tank

2.3 MCRE Design Maturity Overview

The MCRE design utilized to perform this TICAP tabletop exercise is conceptual and, as such, will certainly evolve in the preliminary and final design phases. The purpose of the Conceptual Design Safety Basis for the MCRE is to provide an accounting of various hazards and mitigating strategies to ensure the operation of the MCRE can be performed without unacceptable risk to the health and safety of the public. Within this TICAP tabletop, a description of the design and analyses is provided that currently establishes the overall Safety Basis using simplifying calculations that are assumed to represent limiting conditions/performance of the reactor.

An overview of the MCRE Design is provided in Section 3.1 of this report. Due to the conceptual nature of the design information available, the focus of the current analyses is the Reactor Core System, the Reactor Enclosure System (RXE), and the Primary Coolant System (PCS). Less focus was placed on analyzing the tertiary Heat Rejection System (HRS) and other auxiliary systems. Due to the maturity of the available design information, some assumptions were made, particularly with respect to the Reactor Protection System (RPS) and plant control systems.

2.4 Inputs for the Tabletop Exercise

The prerequisites and inputs for this report are taken directly from the conceptual design for MCRE. Because the MCRE design is in the conceptual design phase, many designs and analyses are not at the level to support a final application; however, the design and analyses that have been performed are extensive and will allow for a meaningful representation of the content of application developed via TICAP that corresponds to this MCRE—the first critical MCFR.

Although these preliminary results, including the LBEs identified and the SSC safety classifications, have not been developed using the quantitative analysis that would be necessary for a Final Safety Analysis Report, the tabletop exercise necessitated the assumption that these results were identified at the conclusion of a full iteration of the LMP approach described in NEI 18-04.

At this time, a preliminary PRA model is under development for MCRE. As such, the LBEs used for this tabletop exercise were selected considering the knowledge of the reactor at the time. The list was informed by considering previous reactor design experience. As the design progresses, the list will include specific events that may be revealed through the use of process hazards analysis methods, such as failure modes and effects analysis and/or hazards and operability studies that will be applied. The maturation of the quantitative PRA for MCRE will also result in a new list of LBEs, required safety functions, and a fully RIPB classification of SSCs. SSC classification will be discussed further within Section 3.1 of this report. Generic analyses, including toolsets and models, will be discussed further within Section 3.2 of this report.

Since the LMP process will continue as the MCRE design evolves, the present analyses address several representative hazard scenarios such as reactivity insertion, loss of heat removal or overcooling, and loss of flow. These scenarios are assumed to be representative events for perturbing the reactor through reactivity change, core inlet temperature change, and core mass flow rate change, respectively. As the design progresses, specific LBEs will be analyzed to ensure adequate reactor performance. In the protected transients, the RPS dominates the transient behaviors, and the inherent reactivity feedbacks play a significant role in maintaining the reactor system in a safe state before the RPS actuation.

3.0 DEMONSTRATION ACTIVITIES

The MCRE design and safety case information in the following subsections of this report provides the context necessary to understand the example SAR content presented in the appendices. The content in the following subsections is a mix of the following information:

- Documentation of assumptions that were made in order to allow for the existing MCRE design and safety case information to be used to support the development of example SAR content
- Information that may belong in portions of a SAR but has not been formally developed based upon the TICAP guidance (e.g., it is not at the appropriate level of detail and/or is not in the appropriate format)
- Information that would not belong in a SAR but is necessary to understand the example SAR content presented in the appendices. In an actual licensing application, this information may be technical details that would reside in a topical report, an internal white paper, and/or another deliverable that may or may not be referenced in the SAR.

3.1 General Plant and Site Description and Overview of the Safety Case

MCRE is a 300 kWth nuclear reactor experiment, fueled with molten NaCl-$PuCl_3$ salt (referred to as fuel salt), that will be operated at the INL ZPPR Test Bed (ZTB) demonstration reactor site. MCRE is a pumped molten salt pool-type reactor with a nitrogen cooling system that removes the fission heat through the reactor vessel wall. The INL ZTB provides the heat rejection system, which has the outside environment as the ultimate heat sink. MCRE has been broken down into 12 systems during the systems engineering process applied during the design, as shown in Figure 3. The primary functions of the systems are listed in Table 3.

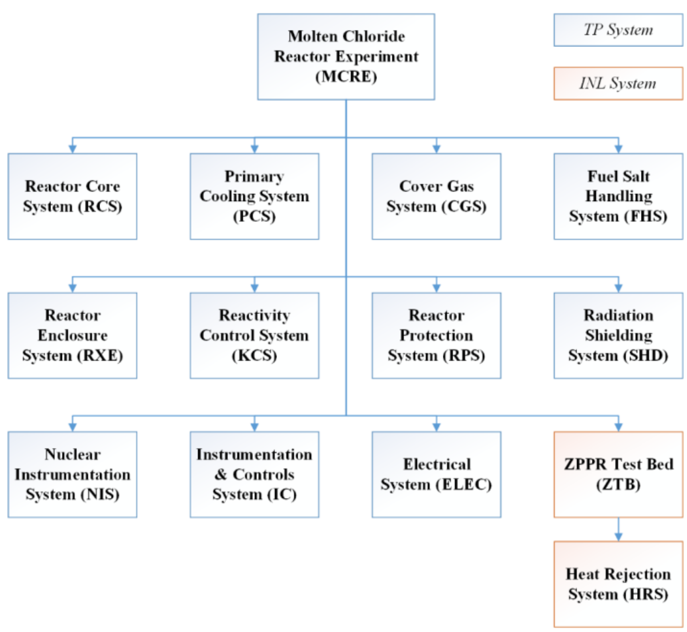

Figure 3. Product Breakdown Structure

Table 3. MCRE Systems and Primary Functions

System	ID	Primary Function(s)
Molten Chloride Reactor Experiment	MCRE	Demonstrate that a fast spectrum, very low βeff, molten salt reactor may be safely controlled; Gather data to validate models
Reactor Core System	RCS	Defines critical geometry for the fuel salt to generate fission heat; Defines fuel salt flow paths to facilitate heat transfer through vessel wall and convergence of fuel salt in active core region
Reactor Enclosure System	RXE	Provides primary pressure boundary for the RCS and fuel salt; Includes finned vessel as primary means of heat removal from RCS and into PCS
Primary Cooling System	PCS	Removes the heat generated in the RCS and transferred through the RXE vessel wall
Heat Rejection System	HRS	Removes the heat from the PCS; Owned by INL ZTB
Cover Gas System	CGS	Maintains cover gas pressure within RCS and FHS; Processes all gases produced during operation, including fission gases
Fuel Salt Handling System	FHS	Melts and transfers flush/fuel salt into and out of the RCS
Reactivity Control System	KCS	Active control of reactivity using control drums; Provides SCRAM capability
Reactor Protection System	RPS	Ensures SCRAM on specific signals
Radiation Shielding System	SHD	Shields workers and equipment from radiation produced during operation
Nuclear Instrumentation System	NIS	Monitors neutron flux and fission power
Instrumentation & Controls System	IC	Controls equipment; Gathers data
Electrical System	ELEC	Provides electrical power; Provides diesel backup power; Provides battery backup power
ZPPR Test Bed	ZTB	Provides heat removal system; Provides site for DOE Authorization

Heat is removed through the vessel wall via a four-circuit nitrogen gas PCS. Each cooling circuit transfers its heat to the HRS, which uses Dowtherm Q as the working fluid; the HRS is provided within the INL ZTB facility as a general-purpose reactor cooling system. The FHS holds, melts, and transfers flush salt and fuel salt in and out of the RCS. Argon cover gas is provided for all salt-containing systems by the CGS. Table 4 summarizes the various working fluids within the MCRE.

Table 4. MCRE Working Fluids

System	Working Fluid
RCS	Fuel salt
CGS	Argon
FHS	Flush salt / fuel salt
PCS	Nitrogen
HRS	Dowtherm Q

3.2 Generic Analyses

3.2.1 Deterministic Safety Analysis Tool

Transient safety analyses for the MCRE were carried out using the RELAP5-3D code, Version 4.3.4.[9] RELAP5-3D has the capability to provide a detailed thermal-hydraulic simulation of the reactor core fuel and primary coolant circuits, as well as the balance-of-plant (BOP). RELAP5-3D is basically a one-dimensional code. An arbitrary arrangement of components in the heat transport systems can be represented in terms of a one-dimensional lumped model. Additional flexibility is provided by the modular design of the code, which makes it easy to modify or replace the treatment for one component without affecting the rest of the model.

The code was originally designed to analyze thermal-hydraulic interactions that occurred during postulated loss-of-coolant accidents in pressurized water reactors. However, as development continued, the code was expanded to include the ability to simulate many of the transient scenarios that might occur in advanced non-water reactors. The code has been successfully used to analyze transients in advanced molten-salt reactors, too, although to a lesser extent.[10],[11] Currently, there are no official benchmark calculations that verify the code's capability for application to molten salt reactors or give a suggested approach for the analysis.

The neutron dynamic model consists of the point-kinetic equations for neutron population and delayed neutron precursor concentrations. The data shared between the thermal-hydraulic and neutron dynamic models are the heat source in the energy conservation are determined by the neutron population as well as the temperature in the energy conservation directly impacting the reactivity feedback in the point-kinetic equations. The built-in point kinetics model of the RELAP5-3D does not include precursor drift terms (discussed in detail in Section 3.2.2) because the code has been applied to only the solid fuel. Thus, the precursor drift effects were indirectly accounted for in the MCRE plant model using a reactivity look-up table that related flow rate changes to changes in the effective delayed neutron fraction.

3.2.2 Reactor Core and Neutronics Model

The reactor kinetics model developed is based on the point kinetics model. The input parameters for the model are comprised of two sections: the point kinetics parameters related to the delayed neutron characteristics and the model parameters for reactivity feedbacks (see Table 5).

Table 5. Neutron Kinetics Parameters Used for Point Kinetics Model

Parameter	Values	
Prompt neutron lifetime, s	1.XXE-6	
Mean neutron generation time, s	1.XXE-6	
Static effective delay neutron fraction (βeff)	0.002XX	
Flowing effective delay neutron fraction (βeff)	0.001XX	
Delayed Neutron Precursors	**βeff ***	**Decay constant, s^{-1}**
Group 1	7.XXE-5	0.01XX
Group 2	5.XXE-4	0.03X
Group 3	3.XXE-4	0.1XX
Group 4	7.XXE-4	0.2XX
Group 5	3.XXE-4	0.8XX
Group 6	1.XXE-4	2.XXX

Figure 4 shows the nodalization of fluid component in the RELAP5-3D model of the MCRE.[12]

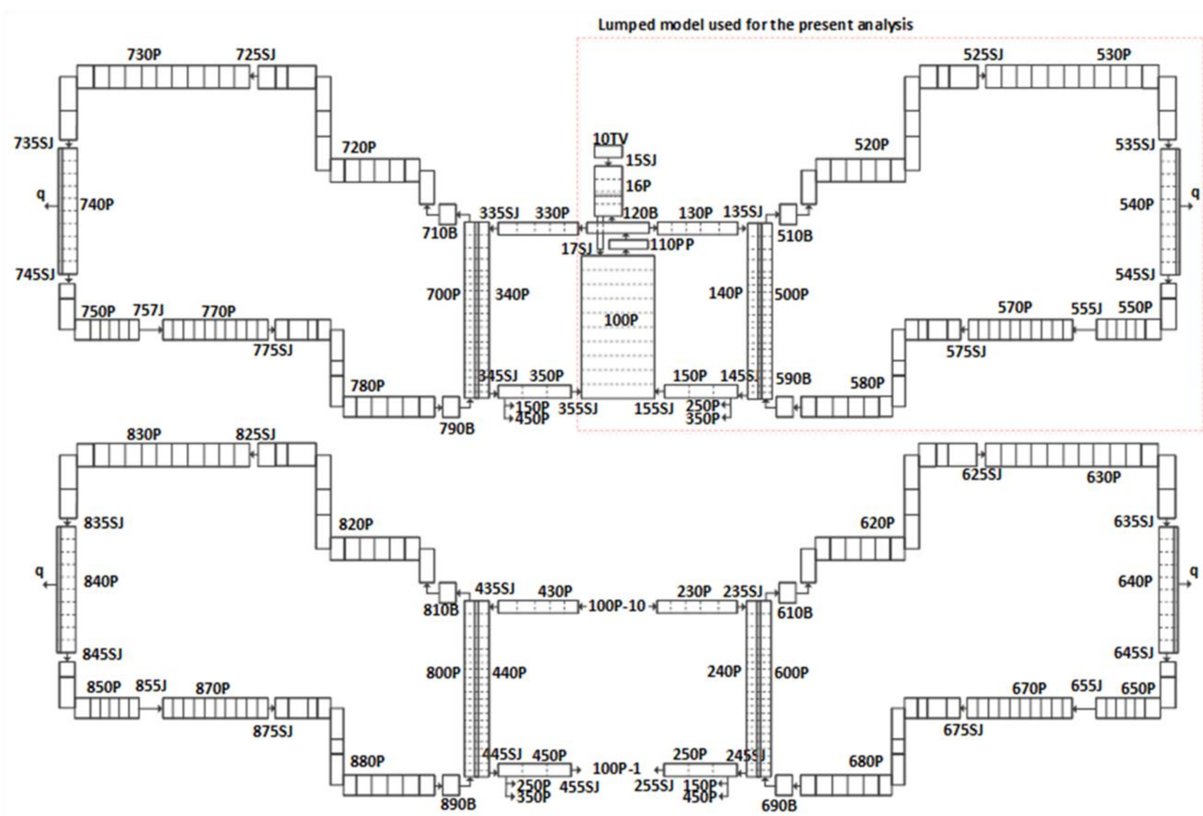

Figure 4. MCRE Nodalization Diagram for RELAP5-3D

* This is the effective delayed neutron fraction.

The fission fragment decay power fraction provided by user-supplied data is shown in Figure 5. To account for uncertainty, a multiplication factor of 1.1 was applied to the decay power curve.[13] The actinide decay power fraction was not accounted for in the analysis.

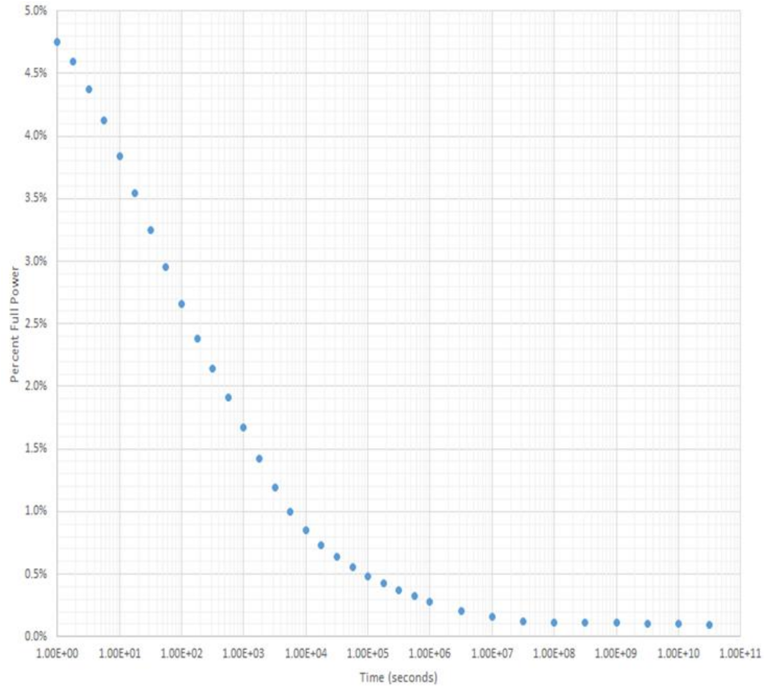

Figure 5. MCRE Fission Product Decay Heat Fraction

The five reactivity feedback mechanisms for the MCRE are due to:

1. Fuel salt Doppler broadening
2. Fuel salt density expansion
3. Radial core structural expansion
4. Axial core structural expansion
5. Variation in the effective delayed neutron fraction (frequently referred to as beta effective, βeff).

For the Doppler and fuel density reactivity feedbacks, the RELAP5-3D built-in models were used.

Other reactivity feedbacks such as the thermal deformation of the reactor structures and advection of delayed neutron precursors are treated separately by developing RELAP5-3D control variable functions. Changes on the time scale of reactor transients in beta effective occur due to advection of the delayed neutron precursors. Reactivity feedback coefficients used for this model are presented in Table 6 and Table 7. As illustrated in Figure 6, the density change as a function of temperature for liquid fuel is significantly higher than for solid fuel.[14] The result is that the reactivity feedback effect due to fuel density (temperature) change is incredibly strong compared to solid-fueled fast and thermal reactors.

Table 6. Reactivity Feedback Coefficients

Reactivity Coefficient	pcm/K	¢/K
Doppler	-0.08 ± 0.04	-0.03X
Density	-3X.X ± 0.1	-1X.XX
Radial structural	+0.7X ± 0.001	0.3X
Axial structural	+0.3X ± 0.001	0.1X
Precursors movement	See Table 7	

Table 7. Reactivity Feedback Due to Variation in Effective Delayed Neutron Fraction

Normalized Mass Flowrate	Difference in Effective Delayed Neutron Fraction $\beta eff(\dot{m}o) - \beta eff(\dot{m}(t))$	Reactivity Change [$]
0	5X.XX	0.3XXX
0.05	3X.X	0.2XXX
0.1	2X.X	0.1XXX
0.2	1X.X	0.1XXX
0.3	1X.X	0.07X
0.4	9.X	0.05XX
0.5	6.X	0.03XX
0.6	4.X	0.02XX
0.7	3.X	0.01XX
0.8	2.X	0.01XX
0.9	1.X	0.005X
1.0	0.0	0
1.1	-1.X	-0.005X
1.2	-1.X	-0.005X
1.3	-2.X	-0.01XX
1.4	-3.X	-0.01XX
1.5	-3.X	-0.01XX

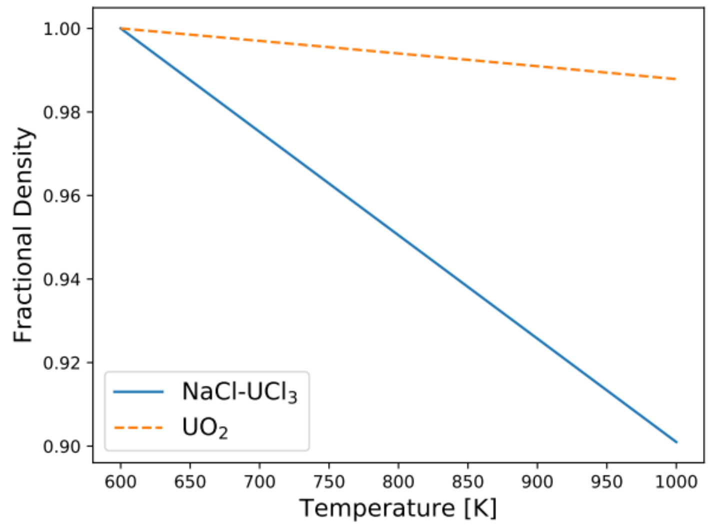

Figure 6. Comparison of Fractional Density Between Solid and Liquid Fuels

The impact of the movement of delayed neutron precursors into and out of the core, and the transit times of the salt components through the RCS, was treated by a simple model in the present analysis. It was assumed that all reactivity changes based on flow rate occur due to differences in flowing beta effective values between a given flow rate and the nominal flow rate. The flowing reactivity inserted at a given flow rate was calculated from a lookup table, as shown in Table 7 above. The flowing beta effective as a function of flow rate was calculated by a modified point kinetics code that explicitly considers the motion of the delayed neutron precursors. The results are plotted as shown in Figure 7. The reactivity change from the delayed neutron precursors drift was encapsulated by the differences of the flowing beta effective.

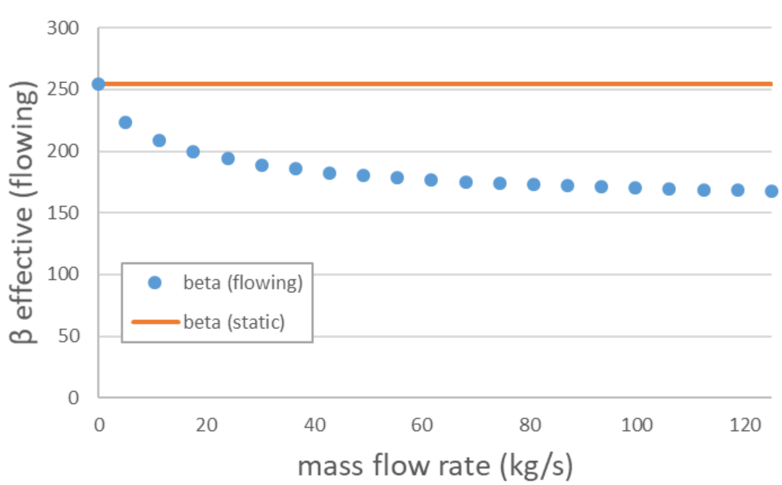

Figure 7. Effective Delayed Neutron Fraction as a Function of Mass Flow Rate

Since the MCRE can operate with little excess reactivity, the reactivity worth required for shutdown does not have to be high. The control drums were assumed to insert the full reactivity into the core from their initial condition upon the reactor trip signal. The integral reactivity worth curves in the case of three out of the four control drums being inserted into the core is depicted in Figure 8. The most reactive drum was assumed to be stuck and not able to participate in the SCRAM. The traveling time of the control drum from the initial position (0 degrees) to the fully inserted position (180 degrees) was assumed to be 1.0 seconds. The actual reactivity insertion rate will depend on the final design of the control drum drive mechanism but is anticipated to be lower than the value assumed in the analysis.

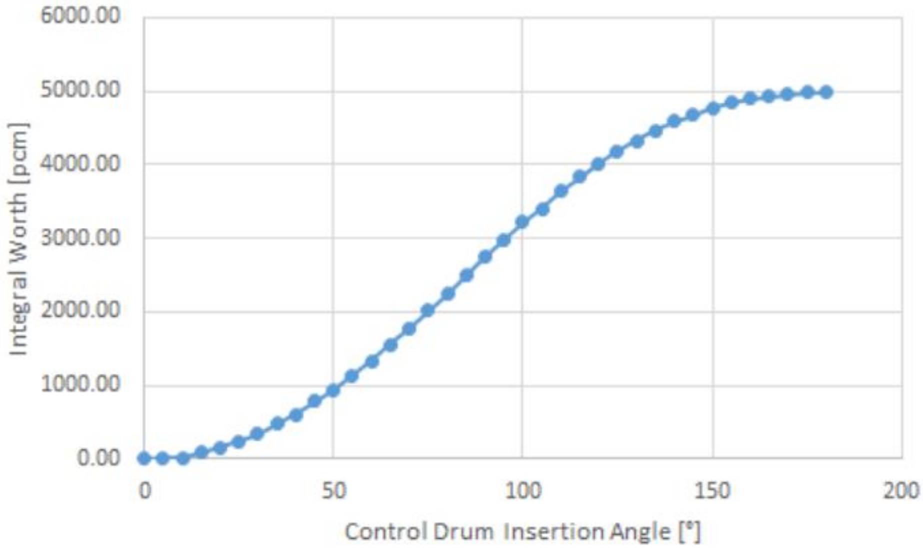

Figure 8. Integral Worth for Three of Four Control Drums Inserted into Core

Protected accident analysis significantly relies on a functional definition of the RPS. The trip thresholds credited in the analysis were the power range high flux (110% RTP), high core-exit fuel salt temperature (7XX ºC), and loss of offsite power. Additional reactor trip parameters are being investigated as part of ongoing design efforts, but they have not yet been specifically identified as necessary. The assumed RPS thresholds and delay times are shown in Table 8. The RPS trip was conservatively assumed to involve the largest possible delay time for sensor detecting, signal processing, and trip brake opening.

Table 8. RPS Thresholds and Response Delay Times Credited in Analysis

Parameter	Threshold	Response Time (s)
Power range high flux	110% RTP	1.0
High core-exit fuel salt temperature	7XXºC	1.0
Loss of offsite power	-	0.0

3.3 Licensing Basis Events

3.3.1 LBE Selection Methodology

At the time of the MCRE tabletop exercise, a PRA model for MCRE was under development; however, a preliminary set of LBEs was selected using a master logic diagram and reviewed by the design team for completeness. The events were selected considering the knowledge of the reactor at the time. The list was further developed by considering previous reactor design experience. As the design progresses, the list will include specific events that may be revealed through the use of failure modes and effects analysis, process hazards analysis, or other similar methods that will be applied.

The master logic diagram considered both dose to the public and dose to workers, but the LBEs of concern for this document pertain only to those events which have some offsite dose potential, not to events that only impact worker dose. From a safety perspective, failures beyond the coolant coils are abstracted as a loss of cooling or excessive cooling at the coils. For the purposes of the TICAP tabletop exercise, it was assumed that the preliminary list of MCRE LBEs displayed in Table 9 was the result of comprehensive and systematic identification of LBEs consistent with the approach described in NEI 18-04.

Table 9. Preliminary List of MCRE LBEs

Initiator (normal) or Group (bold)	LBE Category
High Power Generation	**Group**
Inadvertent positive reactivity insertion by control drums	AOO
Fuel Precipitate Enters the Core	DBE
Cold Fuel Enters the Core	DBE
Sudden Fuel Degassing	BDBE
Premature Criticality (Loading Fuel)	BDBE
100% Heater Actuation at Full Power	AOO
High Fuel Flow	AOO
Loss of Heat Removal	**Group**
Loss of one PCS fan	AOO
Loss of Cooling in One PCS Cooling Coil	AOO
Loss of Power	AOO
PCS Gas Leak	AOO
PCS Gas Break outside bunker	DBE
PCS Gas Break Inside Bunker	DBE
Loss of All PCS Cooling Coils	AOO/DBE (depending on how this system is designed)
Low Fuel Flow	AOO
Mechanical Failures	**Group**
High Cover Gas Pressure	AOO
Low Cover Gas Pressure	AOO
Loss of ZPPR cell cooling	AOO
Pump Seal Failure	DBE
Cover Gas Supply Line Leak	AOO
Cover Gas Supply Line Break or Dewar Leak	DBE

Initiator (normal) or Group (bold)	LBE Category
Cover Gas Purge Line Leak	AOO
Cover Gas Purge Line Break	DBE
Failure of Pump Flange Connection	BDBE
Pump Rotor Rupture	BDBE
Vessel Leak to Cell/Guard Vessel	BDBE
Vessel Leak to PCS Leg	DBE
Leak in Radionuclide Containing Components	AOO
Break in Radionuclide Containing Components	DBA
Overflow Line Leak	AOO
Overflow Line Break	DBE
Freeze Valve Internal Leak	AOO
Fuel Line Leak	AOO
Excessive Heat Removal	**Group**
PCS fan Overspeed	AOO
Overcooling at the Cooling Coil	AOO
Chemical Events	**Group**
Fuel Salt Thermo-Physical Degradation	DBE
Fuel Precipitates on the Vessel Wall	DBE (sub-initiator for other events)
High Vapor Pressure	BDBE
Introduction of contaminants	AOO
Drain and Load Failures	
Inadvertent Criticality	BDBE
Fuel Tank Leak	DBE/BDBE
Loss of Cooling in Drain Tank	BDBE
Heater Over-temperature	AOO
Drain Line Leak	DBA
Overpressure of Pneumatic Fuel Movement System	AOO/DBA
Fuel Storage Failure	
Drop of a fuel cask during loading	DBA
Fuel cask overfill	AOO

3.3.2 LBE Summary

The LBEs discussed in Appendix A of this report were selected to be developed as examples because they both involve the establishment of natural circulation of the fuel salt in the MCRE RCS. These events involve reducing the flow rate of the fuel salt such that the flow is no longer driven by a pump. These flow reduction events will generally reduce the velocity of the fuel salt and thereby decrease the ability of the PCS to remove fission power and decay heat. The events selected are listed as follows:

- Loss of power (AOO)

- Low fuel flow (AOO)

At the time of the MCRE tabletop exercise, quantitative evaluation of the frequencies, consequences, and uncertainties of the LBEs presented has not been completed, and therefore no comparisons will be made to the NEI 18-04 Frequency-Consequence Target curve. Additionally, identification of risk significant LBEs and high consequence BDBEs as defined in NEI 18-04 will not be completed due to the lack of PRA and development of a mechanistic source term model. However, the narratives for the AOOs presented in Appendix A of this report were used to explore the appropriate level of detail expected for these kinds of LBEs based upon the TICAP guidance.

3.4 Integrated Evaluations

This chapter of the SAR was not within the scope of the MCRE TICAP tabletop exercise; thus, no content relating to RIPB integrated evaluations was developed and/or explored.

3.5 Safety Functions, Design Criteria, and SSC Safety Classification

This section of the tabletop report contains information on the safety classification of SSCs that supported the development of the example SAR content in Appendix B (i.e., this information is related to Chapter 5 of the SAR). Because no quantitative PRA model currently exists for MCRE, the preliminary classification of SSCs displayed in Table 10 was assumed to be the output of a rigorous iteration of the RIPB approach described in NEI 18-04.

Table 10. Preliminary Safety Classification of MCRE SSCs

System	System/Component	Safety Classification
RCS	RCS	
	Fuel salt	SR
	Pump shield plug	SR
	Fuel salt pump	SR
	Radial neutron reflector	NSRST
	Lower neutron reflector	NSRST
	Flow guide	SR
	Flow Conditioner	SR
RXE	RXE	
	Reactor vessel (+fins)	SR (+ NSRST)
	Reactor heads	SR
	Reactor skirt	SR
	Thermal shield (outside reactor, inside biological shielding)	NSRST
PCS	PCS	
	Blower	No Special Treatment
	Coil Heat Exchanger	No Special Treatment
	Piping	SR (Inside Biological Shield)

System	System/Component	Safety Classification
		NSRST (Outside Biological Shield)
	Quick-close dampers	SR
	Core heater(s)	NSRST
CGS	CGS	
	Argon Supply Tank	NSRST
	Argon Supply Purifiers (moisture, oxygen hydrocarbon, particulate)	NSRST
	Cover Gas Reactor Vessel Bypass Argon Heater	No Special Treatment
	Cover Gas Fuel Salt Handling Argon Heater	NSRST
	Cover Gas Reactor Pump Argon Heater and line heaters	NSRST
	Cover Gas Reactor Pump Argon Flow Regulating Valve	SR
	Cover Gas Reactor Vessel Argon Pre-Heater and line heaters	NSRST
	Cover Gas Reactor Vessel Argon Flow Regulating Valve	SR
	Cover Gas Reactor Vessel Argon Heater and line heaters	NSRST
	Cover Gas Reactor Vessel Argon Pressure Regulating Valve	SR
	Hydroxide Scrubber Tank and Column	SR
	Hydroxide Scrubber Pump	NSRST
	Hydroxide Scrubber Heaters	NSRST
	Hydroxide Scrubber Particle Filters	SR
	Delay Tank	NSRST
	Delay Tank Pressure Regulating Valve	NSRST
	Carbon Bed Tanks	SR
	Carbon Bed Tanks Pressure Regulating Valve	SR
	Argon Distribution Piping	SR where needed for fuel offload, otherwise NSRST
	Cover Gas (with Fission Product) Piping	SR
	Safety Bottle for Fuel Offload	SR
FHS	FHS	
	Flush salt drain tank	NSRST
	Flush salt drain tank heater	NSRST
	Fuel salt drain tank	SR
	Fuel salt drain tank heater	SR
	Piping	SR
	Freeze valve(s)	SR
	Trace heating	NSRST
KCS	KCS	
	Control drums	SR
	Electric Stepper Motor	NSRST

System	System/Component	Safety Classification
	Position Indicator/Transmitter	SR
	Drive Clutch	NSRST
	Drive Gear(s)	NSRST
	Transmission Gear(s)	NSRST
	Limit Switches	NSRST
	Control Drum Bearings	SR
	Control Drum Support Structure	SR
	Reactivity Control Actuation sub-system Support Structure	SR
	Control Drum Actuator Brake	NSRST
NIS	NIS	
	Source Range Neutron Counters (BF3 chambers or boron-lined proportional counters); 2x	SR
	Wide Range Detectors (Fission Chambers); 2x	NSRST
	Linear Power Monitors	NSRST
	Safety Chambers (Uncompensated Ion Chambers); 3x	SR
RPS	RPS	
	Electromagnetic SCRAM Clutch Assembly	SR
	SCRAM Stop Arm	SR
	Snubber Assembly	NSRST
	SCRAM Spring	SR
	Electrical/I&C Components	SR
	Trip Bistables	SR
SHD	SHD	NSRST
IC	IC	
	Class 1E I&C Control System	SR
ELEC	ELEC	
	Class 1E UPS	SR

3.6 Safety-Related SSC Criteria and Capabilities

At the time of the MCRE tabletop exercise, limited design details were available for specific MCRE SR SSCs. Appendix C of this report displays some example tables that would be included in Chapter 6 of the SAR according to the TICAP guidance. As stated in Section 3.5 of this report, the preliminary SSC classification was assumed to be the output of a comprehensive RIPB evaluation conducted according to the guidance in NEI 18-04.

3.7 NSRST SSC Criteria and Capabilities

Due to the maturity of the available design information, no example content for SAR Chapter 7 was explicitly developed as part of the MCRE tabletop exercise; however, the discussion of

NSRST SSCs and CDC in Appendix B of this report (corresponding to SAR Chapter 5) is related to information that would be presented in SAR Chapter 7.

3.8 Plant Programs

Information on plant programs for MCRE was not available at the time of the MCRE tabletop exercise; thus, content related to SAR Chapter 8 was outside of the exercise's scope.

3.9 Exploration of Generic Molten Salt Reactor Design Criteria within TICAP Affirmative Safety Case

The set of PDC that was identified for MCRE using an RIPB process is displayed in Appendix B of this report as example content for SAR Chapter 5. As an academic exercise, the draft set of Molten Salt Reactor (MSR) PDC being developed for ANS Standard 20.2 were surveyed to better understand the differences between the set of RIPB-derived PDC and the list of PDC derived from the General Design Criteria (GDC) and Advanced Reactor Design Criteria (ARDC).

It is important to note that this activity was only performed as part of the tabletop exercise to develop insights and would not be conducted as part of a designer developing a SAR using the TICAP guidance. The following table illustrates that many of the PDC derived from the GDC and ARDC are important to an affirmative LMP-based safety case that is presented in a SAR developed using the TICAP guidance; however, many of the elements would be identified as plant programs rather than design criteria.

ID	Title	TICAP Team Comment
I	**Overall Requirements**	--
1	Quality Standards and Records	Chapter 8 (Plant Programs)
2	Design Bases for Protection Against Natural Phenomena	Chapter 6 (as Safety-Related Design Criteria [SRDC]
3	Fire Protection	Chapter 8 (Plant Programs)
4	Environmental and Dynamic Effects Design Bases	Chapter 6 (as SRDC)
5	Sharing of Safety-Related SSCs	Not Applicable to MCRE
II	**Protection by Multiple Fission Product Barriers**	--
10	Reactor Design	Assured by use of LMP
11	Reactor Inherent Protection	Required Functional Design Criteria (RFDC) III.1
12	Suppression of Power Oscillations	CDC II.1
13	Instrumentation and Control	Instrumentation is subsidiary to several RSFs as RFDC and in some CDC
14	Fuel Salt System Boundary	RSF I
15	Fuel Salt System Auxiliary System Design	Covered in CDC VI series.
16	Containment Design	RSF V

ID	Title	TICAP Team Comment
17	Electric Power Systems	Subsidiary as a support system in several RFDC and covered in some specific SRDC "Associated Power"
18	Inspection and Testing of Electric Power Systems	Chapter 8 (Plant Programs)
19	Control Room	Worker safety is outside of TICAP scope
III	**Protection and Reactivity Control Systems**	--
20	Protection System Functions	RSF II
21	Protection System Reliability and Testability	Part of RFDC II.3 and Chapter 8 (Plant Programs)
22	Protection System Independence	Assured by use of LMP
23	Protection System Failure Modes	RFDC II.6
24	Separation of Protection & Control Systems	Assured by use of LMP
25	Protection System Requirements for Reactivity Control Malfunctions	RFDC II.4
26	Reactivity Control Redundancy & Capability	Covered separately by RSF II and CDC II series.
27	Reactivity Limits	RFDC II.2 and RFDC III.3
28	Protection Against Anticipated Operational Occurrences	Implied by use of LMP
29	Prevention of Excess Criticality Addition and Prompt Criticality Accident	Essentially contained within RSF I, RSF II, and use of LMP
IV	**Fluid Systems**	--
30	Quality of Fuel Salt System Boundary	Chapter 8 (Plant Programs)
31	Fracture Prevention of Fuel Salt System Boundary	RFDC I.2
32	Inspection Fuel Salt System Boundary	Chapter 8 (Plant Programs)
33	Heat Removal	Covered by RSF I and RSF III
34	Fuel Salt System Cooling	Not applicable for MCRE
35	Inspection of Fuel Salt System Cooling System	Chapter 8 (Plant Programs)
36	Testing of Fuel salt System Cooling System	Chapter 8 (Plant Programs)
37	Containment Cooling	Not applicable for MCRE
38	Inspection of Containment Cooling System(s)	Chapter 8 (Plant Programs)
39	Testing of Containment Cooling System(s)	Chapter 8 (Plant Programs)
40	Containment Atmosphere Cleanup	RFDC V
41	Inspection of Containment Atmosphere Cleanup Systems	Chapter 8 (Plant Programs)

ID	Title	TICAP Team Comment
42	Testing of Containment Atmosphere Cleanup Systems	Chapter 8 (Plant Programs)
43	Structural and Equipment Cooling	
44	Inspection of Structural and Equipment Cooling Systems	Chapter 8 (Plant Programs)
45	Testing of Structural and Equipment Cooling Systems	Chapter 8 (Plant Programs)
V	**Containment**	--
50	Containment Design Basis	RSF V
51	Fracture Prevention of Containment Boundary	Not applicable for MCRE
52	Capability for Containment Leakage Rate Testing	Chapter 8 (Plant Programs)
53	Provisions for Containment Testing and Inspection	Chapter 8 (Plant Programs)
54	Piping Systems Penetrating Containment	Not applicable for MCRE
55	Fuel Salt Boundary Penetration Containment	Not applicable for MCRE
56	Primary Containment Isolation	Not applicable for MCRE
57	Closed System Isolation Valves	Not applicable for MCRE
VI	**Fuel and Radioactivity Control**	--
60	Control of Releases of Radioactive Materials to the Environment	RSF I.3 and CDC VI series
61	Fuel Storage and Handling and Radioactivity Control	RSF IV
62	Prevention of Criticality in Fuel Storage and Handling	RSF IV
63	Monitoring Fuel and Waste Storage	RSF IV
64	Monitoring of Radioactivity Releases	RFDC IV.7 and RFDC V.4
VII	**Salt Systems and Control**	--
70	Reactor Coolant System	Not directly applicable, coolant interface through reactor vessel wall.
71	Fuel Salt and Cover Gas Purity Control	CDC V, cover gas only
72	Salt Heating Systems	RSF IV and CDC III series
73	Cover Gas Line Plugging	RFDC IV.2
74	Salt Leakage Detection and Mitigation	RFDC I.5
75	Quality of the Reactor Coolant Boundary	Chapter 8 (Plant Programs)
76	Rupture Prevention of the Reactor Coolant Boundary	Not applicable
77	Inspection of the Reactor Coolant Boundary	Chapter 8 (Plant Programs)

ID	Title	TICAP Team Comment
78	Fuel Salt System Interfaces	CDC III series
79	Radionuclide Retention Boundary	CDC VI series

4.0 OBSERVATIONS AND CONCLUSIONS

4.1 Observations and Lessons Learned

The MCRE tabletop exercise proved useful in exploring how to narrate the analysis of LBEs. The development of the LBE narratives in Appendix A provided the MCRE team with valuable experience relating key results and insights from the trends predicted by a deterministic model to requirements. Consistent with the LMP approach as described in NEI 18-04, the results may or may not determine the set of safety functions for system components.

The MCRE team noted that a more mature design and computational models with higher accuracy would have increased the value of the tabletop exercise. The analytical results afforded by these models would have allowed deeper exploration regarding how DBE/DBA narratives relate to the RSFs and SR SSCs and provided an example narrative for an LBE other than an AOO.

A great deal of effort was spent understanding how the TICAP guidance was to be applied in developing Design Criteria (Required Functional, Complementary, and Safety Related) that are primarily related to plant functional capabilities in achieving safety, and how they related to traditional Principal Design Criteria that cover a broader range of concerns including maintainability and quality assurance. This eventually resulted in a shared understanding of how the design criteria would be developed and which of the traditional GDC or ARDC are likely to be dispositioned in other areas of an application following the TICAP guidance.

While there was not any significant supporting PRA, the functional assessment with sufficient assumptions still allowed for an understanding of what material would be needed in a more mature application. This includes ensuring adequate identification of functions necessary to reach a safe and stable end state and what equipment performs those functions in LBEs.

4.2 Conclusions

The TICAP tabletop successfully allowed the use of an early design safety basis with a first cut list of LBEs to define the anticipated required safety functions and design criteria and make SSC classifications. A deeper understanding of what content will be necessary in an application using the TICAP format was gained. Development of the Design Criteria, while difficult, eventually led to a shared understanding not just on the topic of developing the design criteria but how to capture some of the functions that would not be directly captured in the LMP process. For example, design features such as natural circulation that may not have a basic event or risk importance are easily identified when crafting narratives and writing what functions are needed to reach a safe end state.

The areas of the TICAP guidance refined by this exercise (as well as the other exercises) will be summarized in the final project report.

5.0 REFERENCES

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Appendix A Draft Content for SAR Chapter 3 – Licensing Basis Events

The content of this Appendix (in black text) is meant to serve as example content that would be displayed in Chapter 3 of a SAR developed using the TICAP guidance. Blue text indicates guidance that has been copied and pasted from the draft TICAP guidance document, while green text represents commentary from the MCRE TICAP team.

A.1. Licensing Basis Event Selection Methodology

As discussed in Section 3.3 of this report, the preliminary set of LBEs identified for MCRE at the conceptual design stage would be refined via execution of the RIPB approach identified in NEI 18-04 before they would be presented as SAR content. Because this preliminary set was assumed to be the final set of LBEs in order to exercise the TICAP guidance, no example content was developed for this section of the SAR as part of this tabletop exercise.

A.2. LBE Summary

A.2.1 Summary Evaluation of AOOs, DBEs, and BDBEs

In this section, a summary of the evaluation of the LBEs is presented. This summary should include:

- Tables with brief word descriptions of the AOOs, DBEs, and BDBEs
- Identification of the radionuclide sources associated with each of the LBEs
- A plot of the frequencies, consequences, and uncertainties of these LBEs with comparison against the NEI 18-04 Frequency-Consequence Target in NEI 18-04 Figure 3-1
- Identification of all risk significant LBEs as defined in NEI 18-04
- Identification of any high consequence BDBEs as defined in NEI 18-04, i.e., those with doses greater than 25 rem
- Definition and success criteria for the reactor-specific safe, stable end states used to define the end states of the LBEs

The word descriptions of the LBE should be described in sufficient detail to indicate the PRA Safety Functions involved in the prevention and mitigation of the LBEs. See Table 5-1 of the LMP LBE report and Sections 5.1 and 5.2 as a general reference.

Safe, stable end states are a key element of the reactor safety case and should be covered in this section. In LWR safety analysis reports, it is generally understood how safe, stable end states are defined in such terms as preventing core damage, maintaining containment integrity, achieving cold shutdown, etc. However, for advanced non-LWRs, the safe, stable end states, including success criteria that are needed to achieve them, need to be defined for the specific technology and design. The plant parameters used to define the end states, e.g., core reactivity, reactor power, fuel temperatures, etc., should be identified.

Table A-1 summarizes the MCRE LBE narratives presented in the following subsections of this chapter, while Table A-2 provides an overview of the end states and relevant inventories of radionuclides for each LBE.

Table A-1. Examples of MCRE LBE Summaries

LBE Designation	LBE Description
Anticipated Operational Occurrences	
AOO-1	Loss of Power initiating event with successful reactor trip, results in a loss of all fans, fuel pump circulation, and chilled water circulation.
AOO-2	The fuel pump slows down initiating event (due to a controller error or similar failure), the reactor is tripped successfully.
Design Basis Events	
DBE-1	A slug of cold/frozen fuel enters the core initiating event, which could be due to overcooling on the PCS side or could even happen alongside the fuel precipitate. A partial blockage of one or more channels would allow that channel to get substantially cooler. As the blockage releases, the cold slug would enter the core.
Beyond Design Basis Events	
BDBE-1	A large portion of the pump rotor fractures from the main body of the pump rotor. This fragment acts as a missile inside the reactor vessel. The reactor also experiences a reactivity insertion due to the loss of flow. The rotor missile could threaten the fuel boundary. It would also generally act as a loose part, and a pump stop transient would occur simultaneously since the broken rotor would not provide much flow.

Table A-2. Overview of End States and Radionuclide Sources for Selected MCRE LBEs

LBE Designation	End State Description	Radionuclide Source(s)
AOO-1	The reactor will shut down shortly after detection of the loss of power. Following the in-vessel shutdown, the emergency fuel offload system will be actuated using only direct current power. The fuel salt will remain in a subcritical configuration within the drain tank and be able to remove heat until power is restored.	Cover gas and fuel salt
AOO-2	As the fuel salt mass flow rate is reduced, the heat removal from the salt also is degraded. This results in slightly elevated temperatures of the fuel salt. The continued heat removal from the nitrogen system drives a temperature gradient, and natural circulation is established in the fuel salt. The reactor transitions to a lower flow rate, with a higher temperature difference across the core to match the flow rate, and the power level returns to the nominal value. Since neither the reactor power level nor the peak temperature values were reached during the transient, the reactor does not SCRAM. With the heat removal systems continuing to operate during this transient, the system can recover without any operator action or reliance on instrumentation to reach the end state.	Cover gas and fuel salt

The example content for this section of the SAR as part of the tabletop exercise was limited to the two tables above. Although the tables do not comprehensively summarize all MCRE LBEs, the content is intended to be representative of the appropriate level of detail for this information. Because a quantitative Frequency-Consequence (F-C) estimate has not been developed for the preliminary list of MCRE LBEs, it was not possible to develop an F-C plot or identify risk-significant LBEs as part of the MCRE tabletop exercise. The MCRE team notes that for a liquid-fueled MSR in particular, the flexibility to define unique end states for each LBE is important.

A.2.2 Summary Evaluation of DBAs

No example content was developed for this section of the SAR as part of the tabletop exercise due to the maturity of the MCRE design and safety case information at the time of the exercise.

A.3. Anticipated Operational Occurrences

This section identifies and describes the plant AOOs that are informed by the PRA event sequence families. AOOs are anticipated event sequences expected to occur one or more times during the life of a nuclear power plant, which may include one or more reactor modules. Event sequences with mean frequencies of 1×10^{-2}/plant-year and greater are classified as AOOs. AOOs take into account the expected response of all SSCs within the plant, regardless of safety classification.

For each AOO, the following information should be provided:

- Narrative of the LBE including definition of the initial plant conditions and plant operating state, radionuclide source (including whether it involves multiple reactors and sources), initiating events covered in the family, response of plant systems, identification of whether or not there is a release, and definition of end state
- Plots of the responses of key plant parameters
- Tables to describe the mechanistic source term if there is a release (or a reference to the source term in Chapter 2)
- The mean and uncertainty percentiles of the estimated frequency and dose

Ideally, the preceding information will be presented for all AOOs. If there are a large number of AOOs, the applicant may elect to provide plant parameter plots, tables of source terms, and uncertainties for only a representative set of AOOs that span the spectrum of safety case challenges and can be demonstrated to be limiting.

This section presents events identified as AOOs.* The plant conditions common to all the AOOs starting from full operating power are discussed and described next.

* According to the TICAP Guidance Document, the LBE selection methodology would be discussed in SAR Section 3.1. Because this section was not developed as part of the MCRE tabletop exercise, see Section 3.3.1 of this report for a discussion of how LBEs were identified.

The thermophysical and calculated nuclear parameters of the fuel are evaluated for fresh fuel, absent of fission products. The RCS and cover gas pressure are at 2XX kPa, and the PCS is slightly above atmospheric pressure at 1XX kPa. The fuel entering the inlet of the active core is at 5XXºC and the outlet at 6XXºC. The nitrogen entering the inlet of the finned heat exchanger is at 7XºC and leaves the heat exchanger at 1XXºC. The RCS pump (RCS-P) is operating at nominal conditions with a mass flow rate of 1XX kg/s, which is split between the four independent fuel coolant channels. The four nitrogen blowers required to cool the four independent coolant channels are operating at nominal conditions providing a mass flow rate of 0.6X kg/s each.

In all these AOOs, it is expected that the reactor can detect a scram signal and trip successfully. Additionally, it is expected that in these events, all radionuclides in the cover gas and fuel salt are retained in the system, and thus, there should be no offsite dose.

The narratives of two AOOs (loss of offsite power and fuel pump failure) were developed as part of the MCRE TICAP tabletop exercise. Because quantitative frequency and consequence estimates have not yet been developed, the narratives focus on the progression of the event from the initiating event to the final end state, including the response of key plant parameters, and do not discuss mechanistic source terms or frequency/dose estimates. Feedback given to the TICAP team based on the development of the following narratives was related to the lack of specificity regarding what constitutes “key parameters.” For a reactor technology without significant precedent (i.e., a liquid-fueled MSR), the MCRE tabletop team did not feel that the TICAP guidance provided clear details on what key parameters should be plotted alongside the narrative, and the suggestion was made to provide additional clarity and/or examples.

A.3.1 Loss of Offsite Power (AOO-1)

The Loss of Offsite Power (LOOP) event considers the scenario where power is lost and no longer available to operate the fuel pump, nitrogen blower, and the heat rejection system. The loss of forced convection of the fuel salt and reduction in heat removal leads to an increase in fuel salt and nitrogen coolant temperatures. The fuel salt temperature is the primary concern. If temperatures are reached that significantly degrade the vessel integrity, the fission product barrier could be compromised. However, it is found that during this event, the increase in temperature of the fuel salt remains well under the upper limit of 7XX ºC. This is primarily due to the reactor scram rapidly reducing fission power upon detection of the LOOP and the presence of the strong negative density temperature feedback, which aids in the reduction of fission power as the temperature increases. Only decay heat power persists but is removed by ambient losses. The fuel salt temperature remains at a slightly elevated temperature and slowly begins to cool as ambient losses and some heat removal by the nitrogen cooling system overcome the decay heat. The nitrogen temperature increases as it continues to remove some heat from the fuel salt even after the blower is no longer providing forced convection. The reactor remains in this sub-critical state, and eventually, the fuel is offloaded to the drain tank. The fuel salt in the drain tank will remain subcritical and is designed to remove heat from the fuel salt with ambient losses and allow for freezing in the tank.

The LOOP event was analyzed using the method described in Section 3.2.* This event represents the simultaneous failure of all pumps and heat removal systems. The initiating event is the loss of normal electrical power to all plant systems. This results in loss of forced convection in the PCS and HRS loops as well as the RCS fuel flow circuits. Simultaneously, heat rejection through the Primary Cooling System-Heat Exchangers (PCS-HXs) decreases to zero as the HRS is unable to perform its function. When the LOOP occurs, sensors monitoring the electric feed will trigger automatic scram of the control drums. There is a 1-second delay between the detection of the LOOP and the beginning of the control drum actuation. The control drum movement occurs automatically and passively. Note that direct current power is credited as available for actuating the control drums if the auto-release system does not function properly.

This event threatens the ability to remove heat from the system, which potentially could lead to an increase in temperature that could affect vessel integrity. When evaluating the severity of this event, an important characteristic to examine is the mass flow rate of the fuel, which is shown in Figure A-1 for the first 100 seconds of the transient.

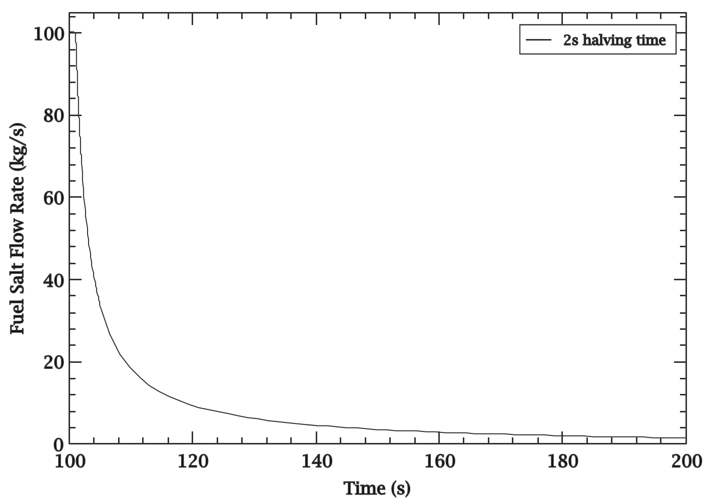

Figure A-1. RCS-P Mass Flow Rate as a Function of Time During the LOOP Transient

The fuel salt temperatures, reactor power, and reactivity feedbacks are shown in Figure A-2, Figure A-3, and Figure A-4, respectively. The increase in fuel salt temperature in Figure A-2 is relatively small, less than 3 K, because reactor shutdown occurs one second after the initiation of the LOOP. Since the temperature transients are mild, there is no significant pressure surge within the vessel due to thermal expansion of the fuel salt. As shown in Figure A-3 and Figure A-4, it is evident that the reactivity worth of the control drums is large enough to shut down the reactor.

* Section 3.2 of this report is provided for context and is not intended to be representative of the content of SAR Chapter 2 developed according to the TICAP Guidance Document.

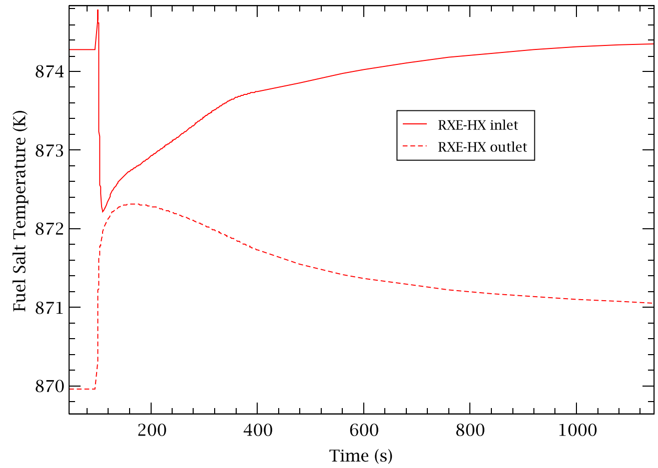

Figure A-2. Temperature of the Fuel Salt at the RXE Heat Exchanger Inlet and Outlet

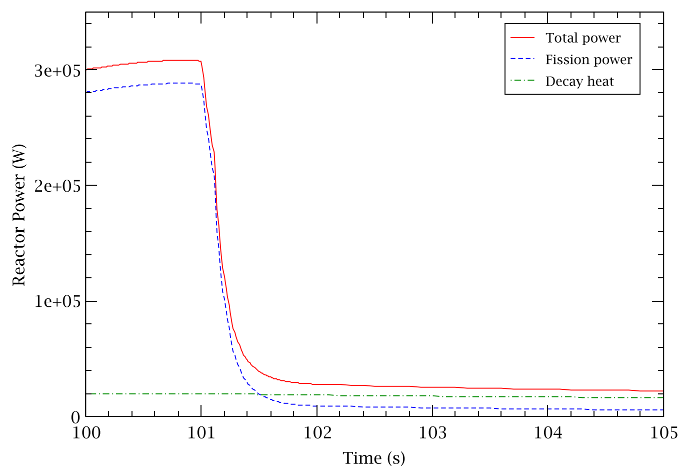

Figure A-3. Reactor Power Response for Loss of Offsite Power

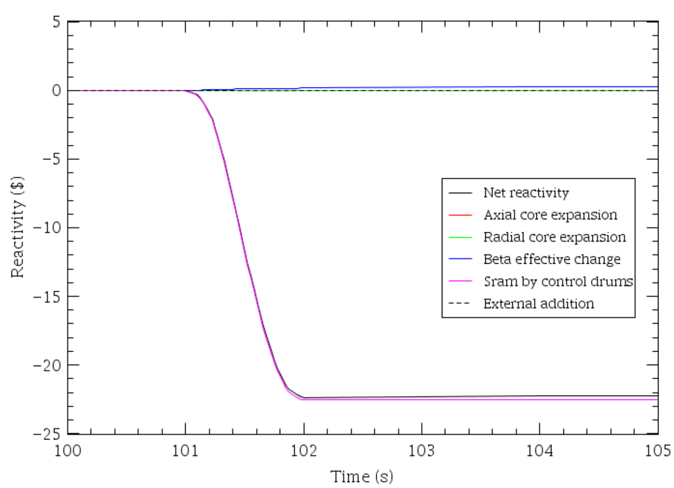

Figure A-4. Reactivity Response for LOOP

A unique feature of a liquid-fueled molten salt reactor is that as the fuel circulates and fission events occur in the fuel, which results in fission products being produced and carried with the fluid. Some of the fission products are grouped together based on the characteristic timescale over which they release delayed neutrons; these groups are referred to as delayed neutron precursors. As the delayed neutron precursors stochastically decay through their radioactive decay chain, they release neutrons with timescales on the order of tenths of seconds to minutes. During transient events where the fuel mass flow rate changes significantly, reactivity insertions may occur from the redistribution of the delayed neutron precursors as the number of delayed neutrons produced in the active core changes depending on the flow rate in the core. Essentially, the fraction of delayed neutrons (β in the point kinetics equations) is no longer fixed and instead varies as a function of the mass flow rate in the system. This effect is illustrated in Figure A-5, which highlights that two different mass flow rates will result in precursors decaying in different positions.

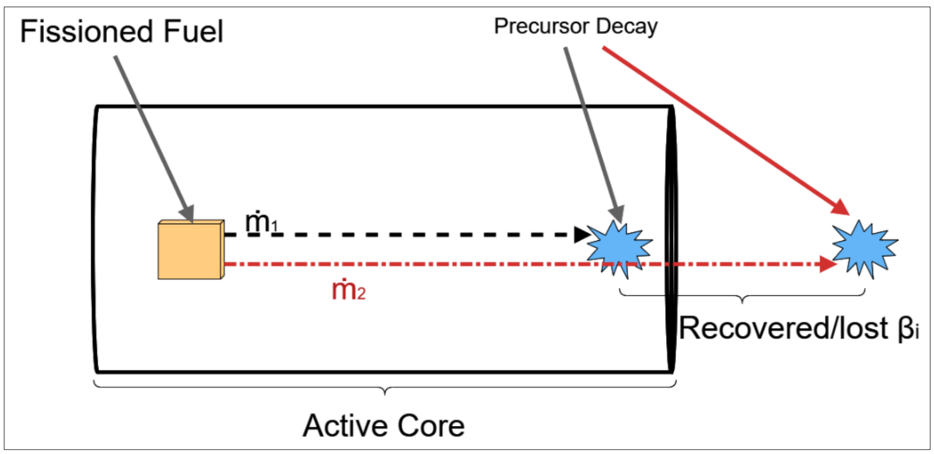

Figure A-5. Illustration that the Delayed Neutron Fraction in the Active Core May Change Depending on the Velocity Through the Core in a Flowing Fuel MSR

In the case of a LOOP event, the fuel salt flow rate in the core decreases, and more delayed neutron precursors decay within the core. Due to the increased production of neutrons from the decay of delayed precursors in the core, there is a small but positive reactivity insertion, as shown in Figure A-4 above. In this event, the maximum reactivity inserted from this precursor movement effect is 5X pcm, as this is the difference between the stationary fuel state and the nominal flowing. However, the magnitude of this positive reactivity insertion is significantly smaller than the control drum scram, and thus its effects are negligible in this transient.

The nitrogen gas temperatures across the RXE Heat Exchanger (RXE-HX) inlet and outlet during the coast down are depicted in Figure A-6. The RXE-HX inlet and outlet temperatures instantaneously decrease upon the reactor core shutdown, but they rapidly increase due to the subsequent reduction of the mass flow rate of the nitrogen gas. Over the remaining time, the natural convection flow in the PCS loops (nitrogen) removes some decay heat from the RCS. The temperature difference between the inlet and out of the RXE-HX remains nearly constant, but eventually, both approach the temperature of the fuel salt as thermal equilibrium between the systems is reached.

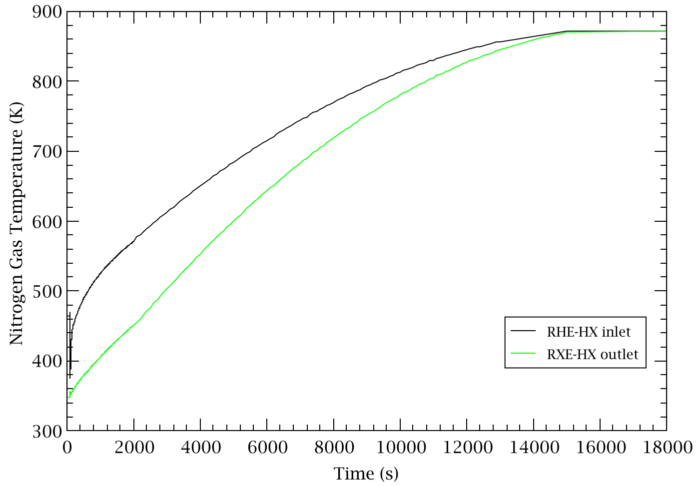

Figure A-6. Nitrogen Temperatures in the PCS for LOOP

The simulated results of this LOOP event show that the fuel temperature remains well under the limit of 9XX K (7XX°C). Based on the mild nature of the temperature increase in the fuel, vessel integrity will be preserved, and cover gas pressure will remain low. Thus, there will be no pathway for fuel salt to leak, and there will be no release of radioactive material during this event. Additionally, no operator action is required, and the only active SSC action required is the automatic scram of the control drums. Following the in-vessel shutdown, the emergency fuel offload system will be actuated using only direct current power. This offload involves melting the freeze valve arrangement between the vessel and the fuel drain tank system, closing any necessary cover gas valves, and providing pneumatic force to offload the fuel from the emergency offload argon bottles. The fuel salt will remain in a subcritical configuration within the drain tank and be able to remove heat until power is restored.

A.3.2 Fuel Pump Failure (AOO-2)

Failure of the fuel pump can occur for a variety of reasons, such as loss of electrical power or mechanical degradation of components. The failure of the fuel pump will result in a loss of forced convection and a reduction in heat removal from the fuel salt. In this event, the PCS and HRS systems continue operating during the transient. The reduction in the mass flow rate of the fuel salt results in an increased temperature difference as the mass flow rate reduces to 5 percent of the nominal value as natural circulation is established. The increase in the average temperature of the fuel salt in the core causes a negative reactivity response, primarily due to the density feedback, and quickly reduces the reactor power. As the system flow rate stabilizes to the natural circulation value, the mean temperature returns to near the starting point, and the power level increases to match. The result is that the system returns to the nominal power level with a lower mass flow rate and larger temperature gradient to match. The temperature gradient in the fuel salt is higher, but the margin remains well under the upper limit of 9XX K. Thus, this event poses no risk to the public or workers on site.

The loss of forced flow event begins from the reactor in full power steady-state operation with conditions described in Section A-1. This event was analyzed using the method described in

Section 3.2.* A complete loss of the forced fuel salt flow may result from a mechanical or electrical failure of the RCS-P or from a fault in the power supply to the pump motor. This event represents the case where loss of forced flow occurs in the fuel salt due to the failure of the RCS-P, but the RXE-HX and PCS-HX are available throughout the transient. Therefore, the HRS and PCS are operating, normally and 300 kW heat is continuously rejected.

Low fuel flow will impact the normal operation of the RCS due to changes in pressure, flow rate, retention of fluid, and/or loss of geometry. Interruption of the fuel salt flow circulation in the RCS causes power and temperature excursions by combined thermal and nuclear mechanisms: a reduction in heat removal from the reactor core and an increase of the effective delayed neutron fraction. If the mass flow of the fuel salt is decreased at any time, the delayed neutron effect induces a positive reactivity feedback, as discussed in Sections A.3.1 and A.3.2.

For the loss of flow events, the major concern is the peak temperatures of the fuel salt and reactor vessel. If the forced flow of the fuel salt is interrupted at high reactor power, the degraded heat removal from the reactor core will cause the fuel temperature to rise in conjunction with the delayed neutron precursor flow effect.

This loss of flow event is driven by the loss of forced circulation of the fuel pump, which is highlighted by showing the reduction in mass flow rate of the fuel salt in Figure A-7. In this case, the mass flow rate decreases to about 5 kg/s (5% of the nominal) as natural circulation is established.

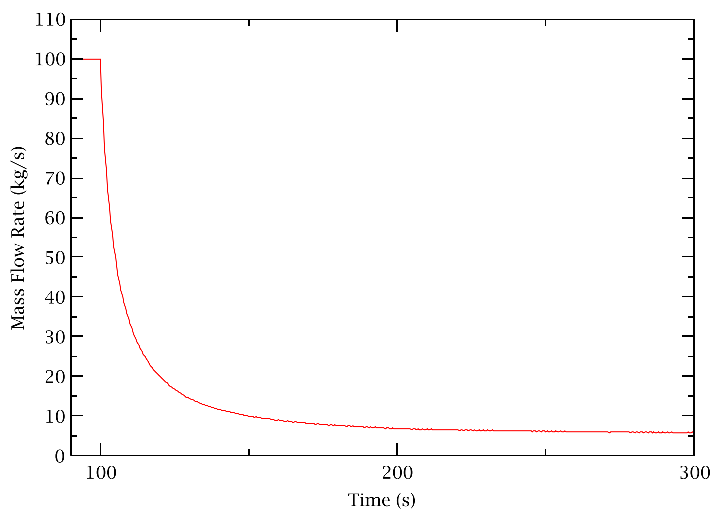

Figure A-7. Fuel Salt Mass Flow Rate

The establishment of natural circulation is driven by the temperature difference in parts of the circuit due to the continual heat removal and the elevation differences between the fuel salt. This is highlighted in Figure A-8, which shows the core fuel salt temperatures up to 30 minutes into the transient. The reduction in heat removal from the core causes the core outlet temperature to

* Section 3.2 of this report is provided for context and is not intended to be representative of the content of SAR Chapter 2 developed according to the TICAP Guidance Document.

rise; however, the mean temperature does not rise considerably. The hottest fuel salt temperature is measured at the location of the RXE-HX inlet as shown in Figure A-9. As the fission heat is continuously rejected through the RXE-HXs and the fuel salt flow decreases, the RXE-HX outlet temperature decreases slightly faster than the inlet temperature.

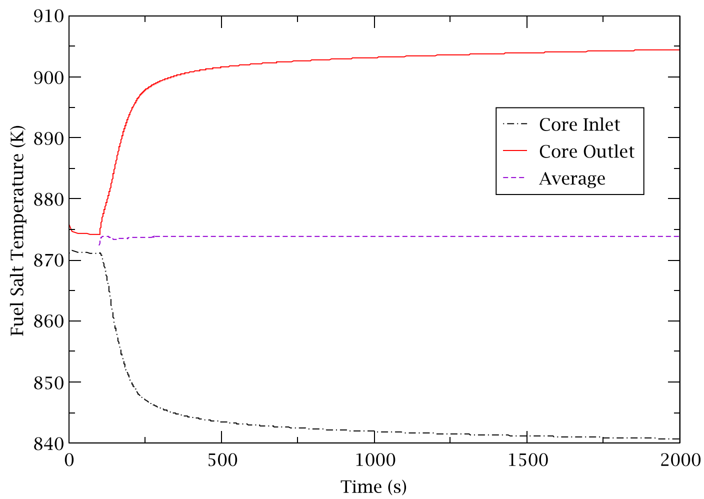

Figure A-8. Fuel Salt Core Temperature Response

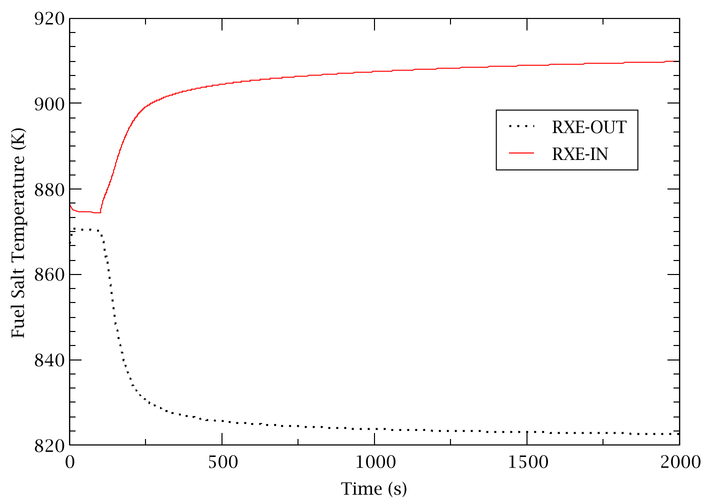

Figure A-9. Fuel Salt Temperatures at the Outlet and Inlet of the Heat Exchanger

The power level and reactivity feedbacks are illustrated in Figure A-10 and Figure A-11, respectively. The average fuel salt temperature increases due to power-to-flow mismatch at the beginning of the transient, which introduces negative reactivity feedbacks. As the mass flow of the fuel salt is decreased, the drift of the delayed neutron precursor provides the most positive reactivity, as additional neutrons are produced in the active core. The axial and radial core expansion reactivities are negligible compared to the other reactivity contributions and act on a much longer time scale. The net reactivity remains nearly zero when the system stabilizes, and the core power comes to a new equilibrium level and has increased to match the heat removal through the PCS-HX.

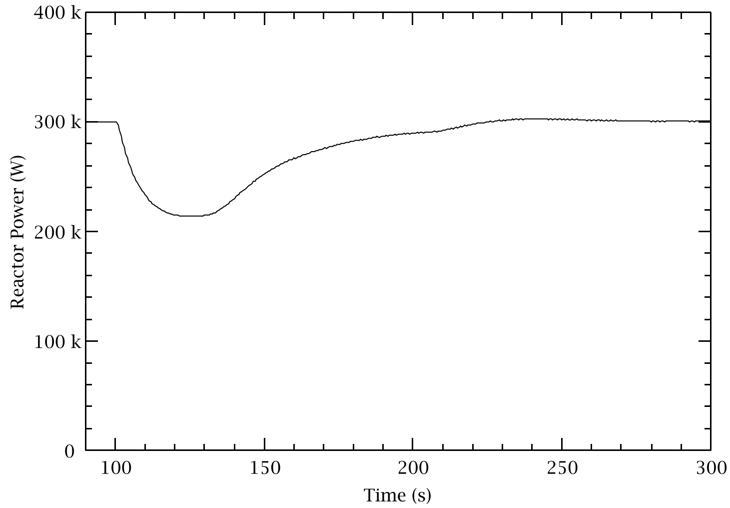

Figure A-10. Power Response for the Loss of Flow Event

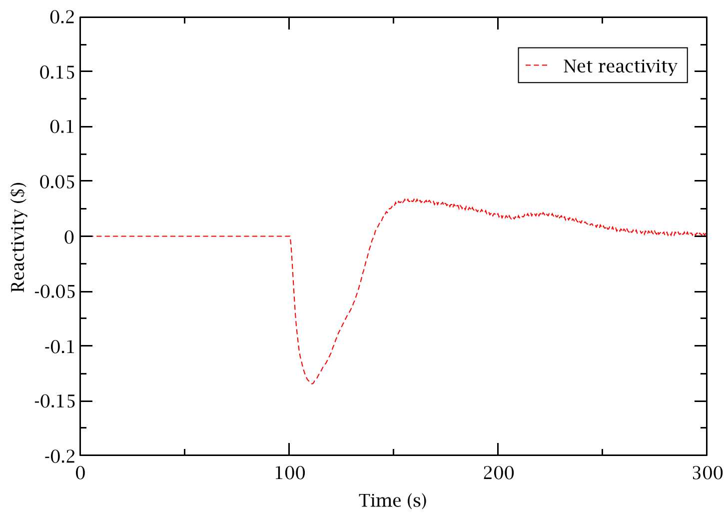

Figure A-11. Reactivity Responses for Loss of Flow Event

The temperature of the fuel salt at the RXE-HX inlet, the hottest part of the fuel salt circuit, increases to about 910 K, which is well below the upper temperature limit of 9XX K (7XXºC) and therefore poses no threat to the vessel integrity. Additionally, the lowest temperature in the fuel at the RXE-HX outlet decreases to about 8XX K, which remains well above the 7XX K (4XXºC) fuel freezing temperature. The establishment of natural circulation results in a new lower mass flow rate to be reached and allows a transition to a new power level close to the starting nominal power. Since neither the reactor power level nor the peak temperature values were reached during the transient, the reactor does not SCRAM. However, the monitoring system is still credited with being available. With the heat removal systems continuing to operate during this transient, the system can recover without any operator action or reliance on instrumentation to reach the end state.

A.4. Design Basis Events

No example content was developed for this section of the SAR as part of the tabletop exercise.

A.5. Beyond Design Basis Events

No example content was developed for this section of the SAR as part of the tabletop exercise.

A.6. Design Basis Accidents

No example content was developed for this section of the SAR as part of the tabletop exercise.

Appendix B Draft Content for SAR Chapter 5 – Safety Functions, Design Criteria, and SSC Safety Classification

The content of this appendix (in black text) is meant to serve as example content that would be displayed in Chapter 5 of a SAR developed using the TICAP guidance. Blue text indicates guidance that has been copied and pasted from the draft TICAP guidance document, while green text represents commentary from the MCRE TICAP team.

B.1. Safety Classification of SSCs

No example content was developed for this section of the SAR as part of the tabletop exercise. Section 3.5 of this report describes how SSCs have been preliminarily classified for MCRE at this stage of design. For the preparation of a SAR, these classifications would be refined via the execution of the RIPB approach described in NEI 18-04; however, to explore the application of the TICAP guidance and develop example SAR content for other sections within Chapter 5, it was assumed that this preliminary SSC classification was the final result.

B.2. Required Safety Functions

This section should present the Required Safety Functions (RSFs), which are the product of applying Step 5a in Figure 3-2 of NEI 18-04. The RSFs are the PSFs that are responsible for successfully mitigating the consequences of all the DBEs inside the F-C Target and for successfully preventing any high consequence BDBEs (i.e., those with doses exceeding 25 rem) from increasing in frequency beyond the F-C Target. A summary level justification for why the reactor-specific RSFs adequately support the FSFs should be included. Examples of RSFs from MHTGR and PRISM are found in the LMP LBE report, and other examples for Xe-100, Kairos FHR, Westinghouse eVinci, MSRE, and PRISM are in the LMP tabletop reports found on the NRC website under Advanced Reactors, Licensing Modernization Project.

The set of RSFs is the minimum set of safety functions necessary to prevent or mitigate consequences from the MCRE initiating events, and through the RIPB approach, the RSFs are used to determine the safety class equipment. For LBEs that do not involve a loss of fuel inventory and that require shutdown, the safety class response is described below.

The reactor is shut down, and circulation of fuel and primary coolant is stopped. Because of the relatively large heat capacity of the system relative to the power produced, the heat changes slowly under decay power conditions. Even a fresh, unburned reactor cools off relatively slowly, allowing shutdown margin to be maintained for a period following shutdown. If the salt were to cool more than 2X°C, the margin to shutdown would no longer be maintained. The temperature of the fuel salt will change less than ten degrees Celsius during the in-vessel shutdown, regardless of operating history.

Once the fuel salt establishes a flow rate via natural circulation, the in-vessel shutdown is indefinitely stable. Various assumptions for this scenario have been considered. The adiabatic case would be expected to heat up less than ten degrees. The no decay power case with conservatively large ambient losses is expected to cool off less than ten degrees. Given these cases, no heat removal or temperature control function is required during the in-vessel shutdown period.

The fuel drain tank system is required to provide a natural circulation heat removal using air cooling and a high surface area to volume ratio, subcritical configuration for the fuel that is drained. This provides an indefinite cold shutdown scenario that requires no further action. It is allowable for fuel to freeze in the drain system, and it is expected that fuel could be melted and transferred back into the reactor following such an emergency offload.

Using this strategy, the following functions will be required to establish acceptable end states and frequencies:

- I. Retain Radionuclides within the Primary System
- II. Reactor Shutdown
- III. Ensure Fuel Remains Liquid
- IV. Fuel Offload
- V. Containment

It is important to recognize that the fuel drain tank performs all three fundamental safety functions once the fuel is offloaded. It acts as a barrier, a cooling system, and provides subcritical geometry for the fuel. Tanks are generally able to be designed as highly reliable components. If for any reason the fuel offload tank is not available, the flush salt drain tank can be used as a standby system to receive the fuel. It is preferred that this not be a normal part of the safety strategy because the fuel would be diluted with the flush salt, making restart difficult.

For fuel or radionuclide leak events, the leak path factor for the ZPPR cell is the only feature necessary to be credited other than isolation already available for offload purposes. Additional means for preventing fuel and fission product release will be implemented. Fuel barrier piping, tanks, and vessels will have secondary (double) walls with leak detection. The design and performance of these secondary barriers may be set by onsite worker dose requirements and/or defense-in-depth measures.

B.3. Required Functional Design Criteria and Principal Design Criteria

Regulations (10 CFR 50.34 or 10 CFR 52.47) require the identification of PDC. For reactors that use the NEI 18-04 methodology, the PDC that flow from the LMP methodology and are needed to support the LMP-based safety case are derived from the RSFs and the RFDC. The identification of RFDC is described in Task 7 in Figure 4-1 of NEI 18-04. Each RFDC constitutes a PDC. There may be additional PDC that cover items outside the scope of the LMP methodology.

This section should present the PDC in terms of the RFDC for each of the RSFs as described in Task 7 of Figure 4-1 in NEI 18-04. These RFDC may be regarded as a decomposition of the RSFs into sub-functions that are necessary and sufficient to support the RSFs. The key elements of the RFDC that should be identified include:

- The success criterion for each of the design specific RSFs
- A breakdown of each RSF into reactor design specific sub-functions that are necessary and sufficient to ensure successful completion of the RSF for all the DBAs. These form a bridge between the RSFs that are defined at a high level and the SRDC.
- An identification of the design-specific inherent or intrinsic reactor characteristics that must be preserved to support the LMP Based Safety Case and are credited in the selection of the SR SSCs

Table B-1 decomposes the RSFs for the MCRE design into RFDC. The RFDC are necessary and sufficient to support the RSFs for MCRE during all identified LBEs; as such, each RFDC constitutes a Principal Design Criterion.

Table B-1. Principal Design Criteria to Support the RSFs of the MCRE Design

Required Safety Function	Required Functional Design Criteria / Principal Design Criteria
I. Retain Radionuclides within the Primary System	1. The primary system boundary shall be designed to reliably retain fuel and other radionuclides under operating, maintenance, testing, and postulated accident conditions. 2. The fuel salt system boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, the probability of rupture is minimized. The design shall reflect consideration of service temperatures, service degradation of material properties, creep, fatigue, stress rupture, and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of coolant chemistry and irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws. (20.2 DC 31) 3. Seals with gas purges such as the pump shaft seal, pump flange seal, or any other identified seal shall be sufficiently functional under normal operating conditions and during postulated accidents to meet release limits. 4. SSCs that support the fuel system boundary, such as the reactor skirt and pipe hangers, shall be designed such that the boundary does not lose support during normal and postulated accident conditions. 5. Instrumentation shall be provided to monitor radiation outside the primary system over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate, to ensure adequate safety.
II. Reactor Shutdown	1. The equipment needed to sense, command, and execute a trip of the control drums, along with any necessary electrical power, shall be designed, fabricated, and operated in such a manner that reactor core shutdown is assured during off-normal conditions. 2. The reactor control drums shall be designed to provide sufficient negative reactivity when tripped following AOOs or DBAs to ensure the reactor is shut down with sufficient margin. 3. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection

Required Safety Function	Required Functional Design Criteria / Principal Design Criteria
	function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. 4. The protection system shall be designed to ensure that specified acceptable radionuclide release design limits are not exceeded during any anticipated operational occurrence accounting for a single malfunction of the reactivity control systems. 5. Instrumentation shall be provided to monitor neutron flux over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate, to ensure adequate safety. 6. The reactor protection system shall be designed to trip the reactor if conditions such as disconnection of the system, loss of energy, or postulated adverse environments are experienced.
III. Ensure Fuel Remains Liquid	1. The reactor shall be designed, fabricated, and operated in such a manner that the inherent nuclear feedback characteristics will ensure that the fuel temperature will stay within an acceptable temperature window where the fuel is liquid with margin to boiling or freezing during power operations and off-normal conditions. Additionally, the reactivity control system(s) shall be designed, fabricated, and operated in such a manner that during insertion of reactivity, the reactor thermal power will not exceed acceptable values. 2. The reactor shall be designed, fabricated, and operated in such a manner that the inherent nuclear feedback characteristics will ensure that the fuel temperature will stay within an acceptable temperature window where the fuel is liquid with margin to boiling or freezing while during power operations and off-normal conditions. 3. The reactor enclosure system and associated reactor internals shall be designed such that the fuel salt will naturally circulate to prevent the formation of hot or cold spots that would lead to local boiling or freezing of the fuel salt during and following off-normal conditions and postulated accidents. 4. The reactivity control system shall be designed in such a manner that the maximum rate and amount of reactivity insertion from a control drum will not result in fuel boiling. 5. Means shall be provided to prevent overcooling of the reactor by the primary cooling system that would cause freezing in a fuel channel during normal and postulated accident conditions. 6. Instrumentation shall be provided to monitor fuel salt temperature over the anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate, to ensure adequate safety, including those variables and systems that can affect the fission process and the integrity of the reactor core, reactor fuel salt boundary, and functional containment. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.
IV. Fuel Offload	1. The equipment needed to sense, command, and execute a fuel offload, along with any necessary electrical power, shall be designed, fabricated, operated, and

Required Safety Function	Required Functional Design Criteria / Principal Design Criteria
	maintained in such a manner that the long-term shutdown of the reactor is assured following off-normal conditions. 2. Redundancy and independence designed into the fuel handling system shall be sufficient to assure that (1) no single active failure results in loss of the ability to move fuel from the reactor vessel to the fuel storage and (2) removal from service of any component does not prevent reliable operation of the fuel offload system. (Modified from RPS DC) 3. The fuel storage and handling systems shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed with a residual heat removal capability that can reliably be performed under normal and postulated accident conditions. (Modified 20.2 DC 61) 4. Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations in normal and postulated accident conditions. (Modified 20.2 DC 62) 5. Appropriate systems shall be provided in the fuel storage system to detect conditions that may result in loss of temperature control capability and excessive radiation levels. (Modified 20.2 DC 63) 6. The piping of the fuel handling system shall be designed to withstand the thermal stresses associated with a fuel offload. If heaters are required to prevent exceeding thermal stress limits, the heaters shall be designed such that no single failure results in a loss of the pipe heating function. 7. Instrumentation shall be provided to monitor temperature, fuel salt level, and radiation in the fuel drain tank over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate, to ensure adequate safety.
V. Containment	1. The reactor containment shall be designed such that a complete loss of reactor fuel through a leak or rupture of the fuel system boundary does not result in an unacceptable off-site dose. 2. Containment and associated systems shall be provided to establish a low-leakage barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. The containment leakage shall be restricted to be less than that needed to meet the acceptable onsite and offsite dose consequence limits, as specified in 10 CFR 50.34 for postulated accidents, including a complete leak of the fuel as a pressurized liquid spill. 3. Systems to control radionuclides which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of reaction products and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained. Each system shall have suitable redundancy in components and features and suitable interconnections, leak detection,

Required Safety Function	Required Functional Design Criteria / Principal Design Criteria
	isolation, and containment capabilities to assure that its safety function can be accomplished, assuming a single failure. 4. Means shall be provided for monitoring the containment atmosphere, effluent discharge paths, and plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

A significant portion of the MCRE TICAP tabletop exercise effort focused on the decomposition of the RSFs into the RFDC in order to identify the PDC in the table above. Through a number of working meetings between the TICAP team and the TerraPower team, an understanding of the expected level of detail and appropriate elements of PDC within the LMP-TICAP framework was developed. During the development of the PDC, the TerraPower team provided the TICAP team with feedback noting the high level of design specificity for PDC (i.e., different reactor designs with similar RSFs may have very different PDC) and the delicate balance between the functional and physical elements of each PDC (i.e., it was sometimes difficult to determine if the PDC should refer to a specific SSC or to a function that would be performed by an SSC).

B.4. Safety-Related SSCs

B.4.1 Selection of SR SSCs

This section presents the technical basis for the selection of SR SSCs, presents the SR SSCs, and identifies the RSFs and PSFs reflected in the LBEs in Section 3.

The first set of tables describes the combinations of SSCs that are provided in the design to fulfill each RSF and identification of whether each set of SSCs is available or not on each of the DBEs. There is one table per RSF. The provisions in the design for alternative ways to perform each RSF is one element of Plant Capability DID. The tables identify which combination of SSCs is selected as SR for each RSF.

An example adapted from the Modular High Temperature Gas-Cooled Reactor (MHTGR) examples for a core heat removal RSF is shown in the following table. Note that the selection of SR SSCs in this example includes SSCs needed to preserve the intrinsic characteristics of the reactor such as power level, power density, shape, and selection of materials that enable the RSF to be fulfilled with the other identified SSCs.

SSC Combinations Capable of Providing Core Heat Removal*	Available for DBE-1?	Available for DBE-2?	...	Available for DBE-N?	Selected as Safety Related?
Reactor Heat Transport System Energy Conversion Area (ECA)	Yes	No	...	No	No
Reactor Shutdown Cooling System Shutdown Cooling Water System (SCWS)	No	Yes	...	No	No
Reactor Reactor Vessel (RV) Reactor Cavity Cooling System (RCCS)	Yes	Yes	...	Yes	Yes
Reactor Reactor Vessel Reactor Building (RB) passive heat sinks	Yes	Yes	...	Yes	No

During the tabletop exercise, it was suggested that the above table was better classified as documentation of the execution of the LMP process rather than results of the LMP process such that it would not be appropriate to include within a SAR. The TerraPower team recommended that a table like the one above would be included in records that could be available to NRC staff upon audit instead of in the SAR content.

B.4.2 SR SSC Summary

A summary table as shown below should be presented that lists all the SR SSCs, the AOOs, DBEs, and BDBEs, and the PSFs responsible for preventing and mitigating each of these LBEs. Given there are multiple RSFs and that each RSF may require the use of multiple SSCs, there will, in general, be multiple SR SSCs. Operator actions that may be necessary to perform any of these functions should be identified as well as the instrumentation and equipment needed to implement those operator actions.

The LBE index numbers in the second column should be keyed to LBE indexes identified in Chapter 3, or alternatively spelled out. For each PSF identified in the last column, the spelled-out function should be listed.

A selection of the MCRE SR SSCs is listed in Table B-2 with the LBEs, LBE type, and the PSF(s) that they perform during each LBE.

Table B-2. Summary of MCRE SR SSCs

SR SSC	PSF	LBEs	LBE Type (AOO, DBE, or BDBE)
Reactor Protection System (RPS)	Provides signal for scram	Inadvertent positive reactivity insertion by control drum	AOO
		Fuel precipitate enters the core	DBE
		Cold fuel enters the core	DBE
		Sudden fuel degassing	BDBE
	Provide trip on loss of power	Loss of offsite power (LOOP)	AOO
Reactivity Control System (KCS)	Insertion of negative reactivity during scram	Inadvertent positive reactivity insertion by control drum	AOO
		Fuel precipitate enters the core	DBE
		Cold fuel enters the core	DBE
		Sudden fuel degassing	BDBE
		Loss of power	AOO
		Overflow line leak	AOO
		Overflow line break	DBE
	Provide indication of control drum position to operator	All LBEs requiring drum actuation	All
Reactor Enclosure System (RXE)	Provide barrier for release of radionuclides	All LBEs	All
Reactor Core System (RCS)	Ensure proper flow in natural circulation conditions	LOOP	AOO
		Fuel pump failure	AOO
		PCS gas leak	AOO
Primary Cooling System (PCS)	Stop coolant flow to prevent salt freezing	Fuel pump failure	AOO
Fuel Salt Handling System (FHS)	Stop fuel flow	Premature criticality (loading fuel)	BDBE
	Offload fuel to subcritical geometry	Loss of power	AOO
		Premature criticality (loading fuel)	BDBE
		Freeze valve internal leak	AOO
		Fuel cask overfill	AOO
Safety Chambers (Uncompensated Ion Chambers)	Provides signal for scram	Inadvertent positive reactivity insertion by control drum	AOO
		Fuel precipitate enters the core	DBE
		Cold fuel enters the core	DBE
		Sudden fuel degassing	BDBE

During the tabletop exercise, the example table from the TICAP guidance was reformatted so that the table more readily indicated that the same SSC would perform the same function for multiple different LBEs. Although the table above is not a comprehensive summary of all MCRE SR SSCs, it provides examples of a given SR SSC that performs the same function during multiple different LBEs and a given SR SSC that performs different functions during different LBEs. The TerraPower team noted that this summary table was a key piece of the LMP-based safety case to link SR SSCs to LBEs through the RSFs.

B.5. Non-Safety-Related with Special Treatments SSCs

For the MCRE tabletop exercise, all NSRST SSCs are grouped together in a single summary table (rather than grouped into those that perform risk-significant functions and those that perform functions necessary for adequate DID) because neither a quantitative estimate of frequencies and consequences nor a formal evaluation of defense-in-depth has been conducted yet. This list of SSCs would be refined and grouped according to the TICAP guidance for presentation in a SAR.

B.5.1 Non-Safety-Related SSCs Performing Risk-Significant Functions

B.5.2 Non-Safety-Related SSCs Performing Safety Functions Necessary for Adequate DID

B.5.3 NSRST SSC Summary

A summary table (example provided below) should be presented that lists all the NSRST SSCs, the AOOs, DBEs, and BDBEs, and the PSFs responsible for preventing and mitigating each of these LBEs. The summary table is a collection of the information in the tables in Sections 5.5.1 and 5.5.2. Operator actions that may be necessary to perform any of these functions should be identified as well as the instrumentation and equipment needed to implement those operator actions.

The LBE index numbers in the second column should be keyed to LBE indexes identified in Section 3, or alternatively spelled out. For each PSF identified in the last column, the spelled-out function should be listed.

The summary Table B-3 lists a selection of MCRE NSRST SSCs and the PSFs that they perform during AOOs, DBEs, and BDBEs, to prevent and mitigate each of these LBEs. This table summarizes the NSRST SSCs discussed in 5.5.1 and 5.5.2. There were no operator actions necessary to perform these NSRST functions.

Table B-3. Summary of MCRE NSRST SSCs

NSRST SSC	PSF	LBEs	LBE Type (AOO, DBE, or BDBE)
Core heater(s)	Provides shutdown margin for in-vessel shutdown	LOOP	AOO
Flush salt drain subsystem (including flush salt drain tank)	Provides redundant means for fuel offload if offloading to fuel drain tank is unsuccessful	LOOP	AOO
Radial and lower neutron reflectors	Passively participate in cooldown of core	Fuel salt pump failure	AOO
Thermal shield	Provides appropriate environmental conditions for SR SSCs	LOOP	AOO
		Fuel salt pump failure	AOO
Argon Supply Tank	Capable of providing gas if SR supply fails	LOOP	AOO
Reactivity Control System (KCS) and associated I&C	Provides normal reactivity control	Fuel precipitate enters the core	DBE
		Cold fuel enters the core	DBE
		Sudden fuel degassing	BDBE
	Provides negative reactivity if scram fails	ATWS	BDBE
Reactor vessel fins (RXE)	Provides means to remove heat from fuel salt in core	AOO + Failure to Offload	BDBE
FHS trace heating	Establish conditions appropriate for fuel drain	Freeze valve internal leak	AOO
		Fuel cask overfill	AOO

Similar to the summary table for the SR SSCs, the MCRE team modified the example table format from the TICAP guidance to more clearly link the SSCs to the PSFs. Although the table above is not a comprehensive summary of all MCRE NSRST SSCs, it provides examples of a given NSRST SSC that performs the same function during multiple different LBEs.

B.6. Complementary Design Criteria for NSRST SSCs

The Complementary Design Criteria for NSRST SSCs are defined in terms of the success criteria for the PRA Safety Functions that are represented in the PRA model to prevent and mitigate the LBEs responsible for the safety classification. For example, a PSF safety function might be "Provide adequate heat removal from the reactor following initiating event X," and the success criterion might be "provide a coolant flow rate of Y kg/sec within Z minutes and maintain maximum fuel temperature less than ZZ." SSCs are classified as NSRST either because the LMP risk significance criteria are met as identified in Section 5.5.1, or the criteria for adequate DID established by the IDP are met as identified in Section 5.5.2. The reliabilities and capabilities that are established in the PRA for the PSFs associated with the SSC trigger the meeting of the risk significance or DID adequacy criteria. These in turn serve to prevent and/or mitigate a specific set of LBEs. Hence the CDC for the NSRST SSCs are directly tied to the success criteria established in the PRA for the PSFs responsible for the SSC classification as NSRST.

These should be presented in tabular form by listing the SSC, the PSF(s) responsible for its safety classification as NSRST, and the design criteria that are necessary and sufficient to meet the PSF. There may be more than one PSF that is associated with the NSRST classification and more than one design criterion for each PSF because the SSC may be represented on multiple LBEs.

The table used to link the NSRST SSCs to the relevant CDC was modified from the suggested format in the TICAP guidance. The example table in the guidance contained a column that displayed the "PSF success criterion" for each PSF; however, the MCRE tabletop team had the opinion that this level of detail was too specific for this table. The alternative organization of the CDC explored by the MCRE tabletop team is closer to the presentation of the PDC, as illustrated in the table below.

A significant amount of effort during the tabletop exercise and discussion between the TICAP team and the TerraPower team focused on the proper level of detail for the CDC (especially in comparison to the PDC). Because there are no analogous functional design criteria between the PSF and the design criteria for the NSRST SSC described in NEI 18-04 (like the RFDC for SR SSC), it was not obvious how SSC-specific the CDC should be. Ultimately, the MCRE team determined that the CDC would be more SSC-specific than the RFDC/PDC. MCRE CDC are shown in Table B-4.

Table B-4. MCRE Complementary Design Criteria (Alternative Table)

Function	Complementary Design Criteria	Associated SSCs
I. Secondary Fuel Barrier	1. The reactor vessel shall be provided with a secondary barrier to accommodate a postulated fuel leaks in the primary system. 2. All fuel handling piping systems shall be provided with duplex piping or be located in a cell that is capable of accommodating postulated fuel leaks. 3. The cell containing the fuel salt drain tank and the flush salt drain tank shall be capable of accommodating postulated fuel leaks.	Guard Vessel Secondary Piping Equipment Cells Primary Piping to Quick Close Dampers
II. Reactor Control	1. The equipment needed to sense, command, and execute motor drive of the control rods, along with any necessary electrical power, shall be designed, fabricated, and operated in such a manner that reactor power can reliably be controlled. 2. The equipment needed to sense, command, and execute fuel flow control, along with any necessary electrical power, shall be designed, fabricated, and operated in such a manner that reactor fuel flow rate can reliably be controlled. 3. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. Adequate radiation protection shall be provided to permit access and occupancy	Reactivity Control System Control Room Associated Electrical Power Supply

Function	Complementary Design Criteria	Associated SSCs
	of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent for the duration of the accident. Adequate habitability measures shall be provided to permit access and occupancy of the control room during normal operations and under accident conditions. Equipment at appropriate locations outside the control room shall be provided with a design capability for shutdown of the reactor, including necessary instrumentation and controls if necessary to maintain the unit in a safe condition during the shutdown.	
III. Ensure Fuel Remains Liquid	1. The reactor vessel heaters and necessary electrical power shall be designed to provide sufficient heat to overcome predicted ambient losses such that fuel can be maintained in a subcritical liquid state in the reactor vessel with consideration for the failure of one heater. 2. The equipment needed to sense, command, and execute primary cooling system flow control, along with any necessary electrical power, shall be designed, fabricated, and operated in such a manner that reactor heat removal can reliably be controlled.	Vessel Heaters Associated Electrical Power Supply
IV. Fuel Offload to Flush Salt Tank	1. The flush salt tank shall have sufficient volume to receive the fuel salt from the reactor vessel when an offload is required. 2. The offload to the flush salt tank shall be possible, accounting for any break or plug in the fuel drain tank or attached lines. 3. Criticality in the flush salt tank shall be prevented by use of a geometrically safe configuration in postulated accident conditions where fuel is offloaded to the flush salt tank. 4. The flush salt tank shall have sufficient heat losses under postulated accident conditions to maintain safe fuel temperatures. 5. Instrumentation shall be provided to monitor salt temperature and salt level in the flush salt tank for accident conditions, as appropriate, to ensure adequate safety.	Flush Salt Tank Flush Salt I&C Associated Electrical Power Supply Cover Gas System
V. Cover Gas Supply	1. The cover gas supply system shall be designed to assure that the purity of the gas supplied, including moisture content, is sufficient to reliably operate fuel salt systems.	Cover Gas System
VI. Minor Radionuclide Retention	1. Systems containing radionuclides whose release is not capable of exceeding offsite dose limits but that pose a hazard to operations shall be designed to reliably retain nuclides during normal operations and anticipated operational occurrences. The design shall reflect consideration of service temperatures, service degradation of material properties, creep, fatigue, stress rupture, and other conditions of the boundary material under operating, maintenance, testing, and anticipated operational occurrences. 2. The off-gas system shall include means to control suitably the release of radioactive materials in gaseous effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. 3. Sufficient holdup capacity shall be provided for retention of effluents containing radioactive materials.	Cover Gas System

Appendix C Draft Content for SAR Chapter 6 – Safety-Related SSC Criteria and Capabilities

The content of this Appendix (in black text) is meant to serve as example content that would be displayed in Chapter 6 of a SAR developed using the TICAP guidance. Blue text indicates guidance that has been copied and pasted from the draft TICAP guidance document, while green text represents commentary from the MCRE TICAP team.

C.1 Design Requirements for Safety-Related SSCs (SRDC)

No example content was developed for this section of the SAR as part of the tabletop exercise.

C.1.1 Design Basis External Hazard Levels

No example content was developed for this section of the SAR as part of the tabletop exercise.

C.1.2 Summary of SRDC

The RFDC are identified in Section 5.3, and the RSFs that they support are identified in Section 5.2. For each of the RFDC, this section should identify a set of SRDC appropriate to the SR SSCs selected to perform the RSFs. These SRDC exclude Special Treatment Requirements, which are separately covered in Section 6.2. The RFDC, which are expressed in the form of functions and involve collections of SSCs and intrinsic capabilities of the reactor, may be viewed as a bridge between the RSF and the SRDC. The SRDC are more detailed requirements for specific SR SSCs in the performance of the RSF functions in specific DBAs. Examples of SRDC that were developed for the MHTGR are found in Appendix A of the LMP SSC Report.

For the SRDC, the following information is presented in tabular form as shown in the table below.

- The first column has the SSC name.
- The second column has a short SSC functional description.
- The third column has the RFDC that the SR SSC supports. Most likely, there is only one RFDC associated with each SR SSC, but if there is more than one, all should be listed. Note that the links from the SR SSCs back to the LBEs that define the RSFs are provided in Chapter 5.
- The fourth column lists the SRDC. There may be more than one SRDC for each SR SSC.

The RFDC and safety-related design requirements (SRDC) for the MCRE SR SSCs are presented in Table C-1.

Table C-1. Summary of MCRE SR SSC RFDC and SRDC

System	SR SSC	Relevant Safety Function or Basis	Applicable RFDC	SRDC
RCS	RCS			
	Fuel salt	The reactor fuel salt carries radionuclides within it that are the primary hazard presented to the public. Maintaining the fuel in a liquid state within the reactor is identified as a required safety function. The thermophysical properties of the fuel salt will dictate heat transfer characteristics important to maintaining fuel within the acceptable window.	III.1 III.2 III.3 III.4 III.5 IV.3	The fuel salt shall be sampled and tested during synthesis to demonstrate acceptable thermophysical properties: Boiling point Freezing point As a function of temperature: Viscosity Density Heat capacity Thermal conductivity
		The reactor fuel salt's nuclear properties are important in assessing the reactivity feedbacks and ensuring subcriticality.	III.1 IV.4	The fuel salt shall be sampled and tested during synthesis to demonstrate acceptable nuclear properties.
	Pump shield plug	Reactor internals ensure proper flow in natural circulation conditions	III.3	The pump shield plug shall be designed such that the natural circulation of the fuel around and above it does not create stagnant regions that would result in fuel or reactor upper head temperature limits being exceeded.
	Fuel salt pump	Barrier function (e.g., shaft and seal) rather than pumping function Note: Pump rotor may or may not be classified as a reactor internal for natural circulation flow.	I.1 I.2 I.3	The portion of the fuel salt pump shaft that passes through the reactor head shall be designed to reliably maintain structural integrity in all off-normal conditions. The fuel salt pump seal shall be designed to prevent the unacceptable leakage of radionuclides during all off-normal conditions. The design of the fuel pump seal shall reflect consideration of service temperatures, service degradation of material properties, creep, fatigue, stress rupture, and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of coolant chemistry, and irradiation on material

System	SR SSC	Relevant Safety Function or Basis	Applicable RFDC	SRDC
				properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.
	Flow guide	Reactor internals ensure proper flow in natural circulation conditions	III.3	The pump shield plug shall be designed such that the natural circulation of the fuel around it does not create stagnant regions that would result in fuel or reactor lower head temperature limits being exceeded.
	Flow conditioner	Reactor internals ensure proper flow in natural circulation conditions	III.3	The effects of the flow conditioner on natural circulation shall be designed such that recirculation and swirl of fuel through the reactor core region does not create stagnant zones that would result in fuel or reactor upper head temperature limits being exceeded.
RXE	RXE			
	Reactor vessel (+fins)	Barrier function, fins are not required to remove sufficient heat for shutdown conditions but can remove heat	I.1 I.2	The reactor vessel shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, the probability of rupture is minimized. The design shall reflect consideration of service temperatures, service degradation of material properties, creep, fatigue, stress rupture, and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of coolant chemistry, and irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.
	Reactor heads	Barrier function and natural circulation flow path	I.1 I.2	The reactor heads shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, the probability of rupture is minimized. The design shall reflect consideration of service temperatures, service degradation of material properties, creep, fatigue, stress rupture, and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and

System	SR SSC	Relevant Safety Function or Basis	Applicable RFDC	SRDC
				the uncertainties in determining (1) material properties, (2) the effects of coolant chemistry, and irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.
	Reactor skirt	Structural support of Safety Class system	I.4	The reactor skirt shall be designed such that the lower reactor head does not exceed stress limits or lose structural support during normal and postulated accident conditions.
PCS	PCS			
	Quick-close dampers	The quick-close dampers are expected to be the same classification as the portion of the PCS piping inside the biological shield. This depends on frequency of vessel leak through and response to other SBEs. The dampers could also be used to prevent overcooling if the PCS blow trip is insufficient.	III.5	The primary cooling system shall be designed to rapidly reduce coolant flow following an in-vessel shutdown to prevent overcooling and possible freezing of reactor fuel with consideration for any single active failure.
CGS	CGS			
	Cover gas reactor pump argon flow regulating valve	Used to maintain the seal (barrier function).	I.3	The argon flow regulating valve leading to the pump flange and shaft seal shall be designed to provide sufficient gas flow to flange seals during normal and postulated accident conditions such that the leakage rate from the fuel system does not exceed acceptable limits for offsite dose.
	Hydroxide scrubber tank and column	Conservatively safety class due to the large potential radionuclide inventory (TBV). Not the controlling hazard for containment function of the ZPPR cell.	I.1	Conservatively included as SR until it can be shown that CDC VI applies. Currently included as "other radionuclide barrier."
	Hydroxide scrubber particle filters	Conservatively safety class due to the large potential radionuclide inventory (TBV). Not the controlling hazard for containment function of the ZPPR cell.	I.1	Conservatively included as SR until it can be shown that CDC VI applies. Currently included as "other radionuclide barrier."
	Carbon bed tanks	Conservatively safety class due to the large potential radionuclide inventory (TBV). Not	I.1	Conservatively included as SR until it can be shown that CDC VI applies. Currently included as "other radionuclide barrier."

System	SR SSC	Relevant Safety Function or Basis	Applicable RFDC	SRDC
		the controlling hazard for containment function of the ZPPR cell.		
	Carbon bed tanks pressure regulating valve	Loss of pressure may cause release of carbon delay bed inventory. Conservatively safety class pending further evaluation.	I.1	Conservatively included as SR until it can be shown that CDC VI applies. Currently included as "other radionuclide barrier."
	Argon distribution piping	Based on classification of other components and functions in the system	I.1 I.2 I.3	Portions of the argon distribution piping that cannot be isolated from the fuel salt are considered part of the fuel salt boundary and shall be designed to reliably maintain structural integrity in all off-normal conditions.
	Cover gas (with fission product) piping	Radionuclide barrier	I.1 I.2 I.3	The cover gas piping is considered part of the fuel salt boundary and shall be designed to reliably maintain structural integrity in all off-normal conditions.
	Safety bottle(s) for fuel offload	Provides gas for fuel offload (so main argon cover gas is safety-significant rather than safety-related)	IV.1 IV.2	The safety bottle(s) shall be designed such that no credible single failure will result in a loss of the ability to provide pressure for fuel offload following postulated accidents. The safety bottle(s) shall either have sufficient redundancy that any component expected to be taken out of service during normal operating conditions would not result in the loss of ability to offload fuel following a postulated accident.
FHS	FHS			
	Fuel salt drain tank	Provides long term "cold shutdown" through critical safe, passively cooled geometry; expected to be used for safety case in an emergency (e.g., LOOP)	IV.1 IV.3 IV.4	The fuel salt drain tank shall be designed to accept fuel following an in-vessel shutdown without exceeding any structural design limits. The fuel salt drain tank shall have sufficient ambient heat losses that fuel and tank temperature limits are not violated by generation of decay heat in the fuel during normal and postulated accident conditions. The fuel salt drain tank shall be designed such that the reactor fuel is subcritical for any credible fuel geometry, including the possibility of external moderation.

System	SR SSC	Relevant Safety Function or Basis	Applicable RFDC	SRDC
	Fuel salt drain tank heater	Not expected to be required to function during an accident but may be needed to establish conditions appropriate for fuel drain; conservatively set as Safety Class for now	IV.5	The fuel drain tank heater shall be designed to provide sufficient heat to keep the fuel liquid with consideration for a single active failure.
	Piping	Radionuclide barrier, if it is ever expected to be fuel contacting in normal operations, AOOs, or DBEs Piping that is exclusively used for flush salt is Safety Significant to perform DID fuel offload function	IV.2 IV.6	The piping of the fuel handling system shall be designed such that no single active failure results in the inability to offload fuel to the drain tank. The piping of the fuel handling system shall be designed to withstand the thermal stresses associated with a fuel offload.
	Freeze valve(s)	Freeze valves that are fuel contacting provide radionuclide barrier function and are required to actuate for fuel offload. Flush salt freeze valves are Safety Significant to perform DID fuel offload function.	IV.1 IV.2	The freeze valve(s) shall be designed to ensure fuel offload capability during normal and postulated accident conditions with consideration of a single active failure.
	Trace heating	Not expected to be required to function during an accident but may be needed to establish conditions appropriate for fuel drain	IV.6	Trace heating will be provided to ensure pipe temperatures prior to fuel offload will prevent thermal shock. Conservatively included until more analysis is complete.
KCS	KCS			
	Control drums Control drum bearings Control drum support structure	Reactivity Control Function	II.1 II.2 II.3	The control drums shall be designed, fabricated, and operated in such a manner that reactor core shutdown is assured during off-normal conditions. The reactor control drums shall be designed to provide sufficient negative reactivity when tripped following AOOs or DBAs to ensure the reactor is shut down with sufficient margin accounting for a single stuck drum.
	Position indicator/ transmitter	Provides operational information of SR system	II.1	The drum position indicator shall be designed to assure drums are inserted following off-normal conditions.

System	SR SSC	Relevant Safety Function or Basis	Applicable RFDC	SRDC
NIS	NIS			
	Source range neutron counters (BF3 chambers or boron-lined proportional counters); 2x	Provides SR signal at low power range	II.1 II.5	Instrumentation shall be provided to monitor neutron flux over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate, to ensure adequate safety.
	Safety chambers (uncompensated ion chambers); 3x	Provides SR signal	II.1 II.5	Instrumentation shall be provided to monitor neutron flux over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate, to ensure adequate safety.
	Detector preamplifiers	Participates in attached function	II.1 II.5	The detector preamplifiers shall be designed to ensure there is sufficient signal produced for a trip and to detect reactor shutdown during normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate, to ensure adequate safety.
	Digital processing units Power/signal cables	Provides SR signal	II.3	Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated.
	Cabinets	Provides SR signal	II.6	The NIS breaker cabinets shall be designed to trip the reactor upon a loss of power.
	Power supplies Amplifiers	Provides power for SR system	II.1	The NIS power supplies and amplifiers shall be designed, fabricated, and operated in such a manner that reactor core shutdown is assured during off-normal conditions.
RPS	RPS			

System	SR SSC	Relevant Safety Function or Basis	Applicable RFDC	SRDC
	Electro-magnetic SCRAM clutch assembly SCRAM stop arm SCRAM spring	Mechanical SCRAM function	II.1	The equipment needed to execute a trip of the control drums, along with any necessary electrical power, shall be designed, fabricated, and operated in such a manner that reactor core shutdown is assured during off-normal conditions.
	Electrical/ I&C components	Provides SR signal	II.3 II.4 II.5	The reactor protection system shall be designed to trip the reactor if conditions such as disconnection of the system, loss of energy, or postulated adverse environments are experienced. The protection system shall be designed to ensure that specified acceptable radionuclide release design limits are not exceeded during any anticipated operational occurrence accounting for a single malfunction of the reactivity control systems. Instrumentation shall be provided to monitor neutron flux over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate, to ensure adequate safety.
IC	IC			
	Class 1E I&C control system	FSH Flush Salt Tank Heating Controls FSH Fuel Salt Tank Heating Controls FSH Freeze Valve Controls CGS Scrubber Controls	II.5 III.6 IV.1 IIV.7	The equipment needed to sense, command, and execute the required safety functions, along with any necessary electrical power, shall be designed, fabricated, operated, and maintained in such a manner that the long-term shutdown of the reactor is assured following off-normal conditions. Instrumentation shall be provided to monitor fuel salt temperature over the anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate, to ensure adequate safety, including those variables and systems that can affect the fission process and the integrity of the reactor.
ELEC	ELEC			

System	SR SSC	Relevant Safety Function or Basis	Applicable RFDC	SRDC
	Class 1E UPS	RPS Electrical Power NIS Electrical Power FSH Electrical Power (Flush/Fuel Salt Tank Heating/Drain Lines/Freeze Valves) CGS Electrical Power (Scrubber Recirculating Pump Power) IC Electrical Power	I.1 I.3	The power supplies to SR components shall be designed in a manner that reactor core shutdown is reliably achieved. The power supplies shall be designed such that no single failure or removal of component from service results in a loss of the supported safety function.

The above table was developed based upon an example table in the TICAP guidance. One insight gained during the preparation of the table during the MCRE tabletop exercise was that multiple RFDC can apply to a single SR SSC; this possibility was not explicitly stated in the TICAP guidance.

C.1.3 Summary of DBEHL-related Requirements for NSR SSCs

No example content was developed for this section of the SAR as part of the tabletop exercise.

C.2 Special Treatment Requirements for SR SSCs

No example content was developed for this section of the SAR as part of the tabletop exercise.

C.3 System Descriptions for SR SSCs

No example content was developed for this section of the SAR as part of the tabletop exercise.