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

X-energy Xe-100
TICAP Tabletop Exercise Report

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Disclaimer

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 X Energy, LLC (X-energy) to explore the application of the draft TICAP guidance to the safety case for the Xe-100 reactor design. Example content for SAR Chapter 2, Methodologies and Analyses, and SAR Chapter 8, Plant Programs, were developed and feedback from the development of this content informed revisions to the TICAP guidance. In addition to the example SAR content, this report provides additional context about the Xe-100 design and safety case and documents the major lessons learned about the TICAP guidance during the Xe-100 tabletop exercise.

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List of Abbreviations

AOO	Anticipated Operational Occurrence
BDBE	Beyond Design Basis Event
COL	Combined License
DBE	Design Basis Event
DID	defense-in-depth
DOE	Department of Energy
EAB	Exclusion Area Boundary
F-C Target	Frequency-Consequence Target
FSF	Fundamental Safety Function
FW	Feedwater
HPB	Helium Pressure Boundary
HTGR	High-temperature gas-cooled reactor
IE	Initiating Event
INL	Idaho National Laboratory
LBE	Licensing Basis Event
LMP	Licensing Modernization Project
LWR	light water reactor
MHTGR	General Atomics Modular HTGR
NEI	Nuclear Energy Institute
NGNP	Next Generation Nuclear Plant
non-LWR	non-light water reactor
NRC	Nuclear Regulatory Commission
NSRST	Non-Safety-Related with Special Treatment
PBMR	Pebble Bed Modular Reactor
PRA	Probabilistic Risk Assessment
PSRV	Pressurizer Safety Relief Valve
RB	Reactor Building
RCCS	Reactor Cavity Cooling System
RFDC	Required Functional Design Criteria
RIPB	risk-informed and performance-based
RSF	Required Safety Function
SAR	Safety Analysis Report
SG	Steam Generator
SR	Safety-Related
SSCs	Structures, Systems, and Components
SU/SD	Start-up/shutdown
TICAP	Technology Inclusive Content of Application Project
TRISO	Tristructural isotropic
UCO	Uranium oxycarbide

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, the 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 Xe-100 design in coordination with X-Energy, LLC (X-energy). The Xe-100 tabletop exercise explored the development of example SAR content for Chapter 2, Generic Analyses, and Chapter 8, Plant Programs, 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 Xe-100 tabletop exercise and this report.

Table 1. Relationship of Relevant LMP and TICAP Documents to Xe-100 Tabletop Exercise

Document	Relationship to Xe-100 tabletop exercise	Ref.
NEI 18-04	The LMP approach documented in NEI 18-04 (and endorsed by NRC in Regulatory Guide 1.233) is the basis for the approach used by X-energy to develop an RIPB safety case for the Xe-100 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.	[1]
LMP reports	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 (LBEs); (3) safety classification and performance criteria for SSCs; and (4) RIPB evaluation of Defense-in-Depth (DID). 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.	[4], [5], [6], [7]
High-Temperature Gas-Cooled Pebble Bed Reactor LMP demonstration report	An LMP demonstration activity was conducted in 2018 using the Xe-100 design at that time as a baseline to exercise portions of the LMP approach. Although the design and safety case has changed since the publication of the LMP demonstration report, relevant design and safety information has been incorporated into this report to provide context to the reader.	[2]
TICAP Guidance Document	The TICAP guidance document was used to identify the content, structure, and level of detail of the example SAR content developed for the Xe-100 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 publicly available.	[8]

1.4 Xe-100 Tabletop Exercise Objectives, Scope, and Deliverables

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 February 2021 between the TICAP team and the X-energy tabletop team. During these working meetings, the focus was on discussing the example SAR content as it was developed and gathering feedback from the X-energy 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 Xe-100 tabletop exercise. Example SAR content that would belong in Chapter 2, Methodologies and Analyses, and content that would belong in Chapter 8, Plant Programs, is displayed in the appendices of this report. Additionally, some concepts that relate to a SAR developed using the TICAP guidance, such as the identification of Required Functional Design Criteria (RFDC) that are RIPB design-specific Principal Design Criteria, are discussed in the body of the report, although the format and structure are not representative of how this information would look in a SAR.

The third objective of the Xe-100 tabletop exercise was achieved by a working meeting between the TICAP team and the X-energy tabletop team on February 3, 2021, which was observed by NRC and INL staff. Example SAR content in Chapter 2 and Chapter 8 was discussed during the meeting. Major feedback items were also discussed to provide 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.

1.5 Report Organization

The remainder of this report is focused exclusively on the Xe-100 TICAP tabletop exercise. The following section (Section 2) sets the context for the Xe-100 tabletop exercise and provides information about the maturity of the design and safety case information used during the exercise. Section 3 contains technical information to support understanding of the draft SAR content that is shown in the appendices. The information in Section 3 includes assumptions that were necessary to make in order to develop the example SAR content, content that is too detailed for the example SAR content but provides helpful context and other discussions that belong in a SAR chapter outside of the scope of the Xe-100 exercise (i.e., other than Chapter 2 or Chapter 8). Observations, experiences, and lessons learned from the activity are presented in Section 4 from the developer's perspective. References are given in Section 5. Finally, Appendix A displays the draft SAR content developed for Chapter 2, Methodologies and Analyses, and Appendix B displays the draft SAR content developed for Chapter 8, Plant Programs.

[*] https://www.nrc.gov/docs/ML2105/ML21050A114.pdf

2.0 DEMONSTRATION OVERVIEW

2.1 Summary of Demonstration Activities

The Xe-100 TICAP tabletop exercise took place between November 2020 and February 2021. The focus of the exercise was to develop example SAR content using the draft TICAP guidance for two chapters (i.e., SAR Chapter 2 and SAR Chapter 8) to provide feedback to the TICAP team regarding potential improvements to the draft guidance and to produce example SAR content for the benefit of future guidance users. The tabletop team largely relied upon the conceptual Xe-100 design, safety analyses, and licensing documentation that have been generated to date to produce the example SAR content. Additional information was drawn from the 2018 LMP tabletop report for the Xe-100 design.

The tabletop exercise included regular virtual working meetings between the TICAP team and the X-energy team to discuss the example SAR content as it was being developed, as well as potential areas for improvement identified by the X-energy team and proposed resolutions by the TICAP team. A tabletop meeting, including observation by NRC and INL staff, was held virtually on February 3, 2021. The focus of the meeting was to discuss the example SAR content (presented in the appendices of this report) and discuss the major feedback and lessons learned during the exercise. The meeting allowed for direct feedback from the NRC and INL observers regarding how the content (and tabletop exercises) met their expectations and where there were deviations regarding areas including interfaces between the information in the TICAP portion of an advanced reactor licensing application and information in other portions of the application.

2.2 Xe-100 Design Status

The Xe-100 is a 200 MWt, 80 MWe per unit high-temperature gas-cooled reactor (HTGR) with a pebble-bed core. This modular design is an evolution of past HTGR technology that uses tristructural isotropic- (TRISO-) coated particle fuel in the form of billiard ball-sized pebbles, uses helium as a means of heat transport, and produces high-quality steam in a helical-coil steam generator for use by a turbogenerator for electricity production, process heat, and cogeneration arrangements. A standard offering is a four-unit plant with some common SSCs for maintenance and operations. Each unit has a single reactor-steam generator pair, reactor and auxiliary buildings making up a Nuclear Island, and a Conventional Island that houses Non-Safety-Related with no Special Treatment power generation SSCs, separated from the Nuclear Island by isolation valves.

At the time of TICAP tabletop planning, development of the Xe-100 Nuclear Island SSCs was at the end of the conceptual design phase, with readiness reviews ongoing for transition to preliminary/basic design. This phase transition included the development of functional requirements, preliminary SSC process flow and information documentation, drawings, layouts, and performance requirements, as well as performance analysis of integrated systems as part of both design and safety assessments. These activities were informed by the iterative development processes described in NEI 18-04 and led to preliminary SSC classifications, a Phase 0 PRA, initial frequency-consequence curve results, and preliminary LBE event sequences and categorizations.

Since the Xe-100 is an evolutionary HTGR design, it was possible to build upon the design, analyses, and risk-insights of past HTGR technologies like the General Atomics modular HTGR (MHTGR), the Pebble Bed Modular Reactor (PBMR), and proposed Next Generation Nuclear Plant (NGNP) designs. This allowed the TICAP tabletop team to draw from past MHTGR licensing documents (e.g., NUREG-1338) and LMP-related activities (e.g., significant MHTGR lessons learned found throughout NEI 18-04 and supporting LMP reports).

2.3 Prerequisites and Inputs for the Tabletop Exercise

The initial planning for X-energy participation in the TICAP tabletop demonstrations identified two challenges to providing sufficient prerequisite information and support. First, the X-energy team was in the midst of completing an iteration of LMP-based information to support customer requests that resulted in the need to prioritize certain resources during the tabletop phase. Some of this information is provided herein but was not necessarily used to support the NRC and INL staff observations. Second, the down-selection of X-energy as a DOE Advanced Reactor Demonstration Program participant occurred just before the tabletop phase began. The X-energy team determined that some original tabletop scope would not be accomplishable based on new schedule commitment conflicts.

X-energy provided the following information in support of deliberations on the scope, level of detail, organization, and type of information in an LMP-based safety case and Chapter 2 and 8 content:

- Preliminary information on the Xe-100 PRA, mechanistic source term, and transient analysis methodologies
- The existing LBE list, required safety function development, and SSC classification tables from the 2018 LMP tabletop exercise
- Updated SSC classification tables and preliminary special treatment considerations that informed the information required to describe plant programs
- An overall technology description for common awareness and understanding of the plant design
- Insights from the PRA and Required Safety Function (RSF) decomposition from the MHTGR design

X-energy provided context around the information generated and provided for use in the TICAP tabletop that, while reflective of the overall design effort, was selected and applied for the purpose of exercising the TICAP guidance for a notional Part 52 Combined License application and should not be construed as representative of any other application type or project. While future applications may use similar information and structure, the TICAP tabletop example content was developed primarily to exercise and improve the TICAP guidance document.

3.0 DEMONSTRATION ACTIVITIES

The information in the following subsections is intended to provide context about the Xe-100 design and safety case to maximize the usefulness of the example SAR content displayed in Appendix A and Appendix B. Although some of the content in Section 3 of this report relates to content that would be covered in the portion of a SAR that is covered by the TICAP guidance, the information in Section 3 of this report differs from the information in the appendices in that it is not intended to be representative of the format, structure, or level of detail of a SAR prepared using the TICAP guidance.

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

The Xe-100 reactor design was described in principle in the 2018 LMP report, an updated summary of which follows. X-energy finalized a technical report, submitted in April 2021, to aid NRC staff awareness during topical report reviews that more fully describes the Xe-100 reactor technology.

A single Xe-100 reactor is designed to produce 200 MWt in a pebble-bed core. Circulating helium transfers heat to a single steam generator with dual helical coils designed to produce high-grade, superheated steam at 565°C and 16.5 MPa. This steam may be used to produce a maximum 80 MWe by turbogenerator, high-temperature process heat for industrial markets, or multiple energy products in a cogeneration arrangement. The standard design Xe-100 plant deploys four identical 200-MWt units (four-pack), each consisting of a Nuclear Island containing the reactor/steam generator coupled to a Conventional Island, as shown in Figure 2**Error! Reference source not found.**.

Figure 2. Xe-100 Deployment – Four-Unit Plant

The Reactor Pressure Vessel and Steam Generator (SG) arrangement is shown in Figure 3. Each reactor on a four-pack plant can operate independently of the others, with the main shared operational facilities being the control room and high-energy interim storage facility for used fuel canisters. Each unit is designed for safe and secure operation during concurrent construction of additional units at the site.

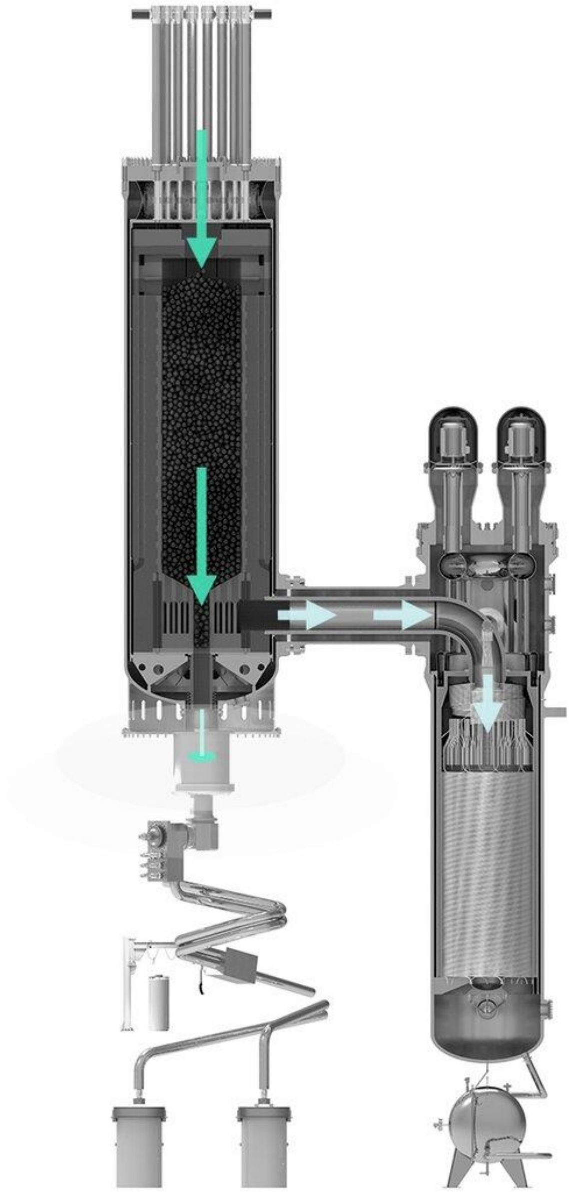

Figure 3. Xe-100 Reactor Pressure Vessel and Steam Generator Arrangement

The reactor uses helium as the primary-side heat transport fluid to transfer heat to the power conversion systems. Helium inlet (260°C) and outlet (750°C) temperatures provide significant material limit margins for structural graphite and metallic support structures. The inlet temperature of 260°C ensures that the Reactor Pressure Vessel and Core Barrel can be designed to existing ASME codes and standards without the need for additional code cases.

The reactor's primary circuit circulates helium through the reactor, then through the helical coil steam generator via a double-walled cross-vessel (see Figure 3). The SG contains two independent tube bundles (independent dual-loop helical coils) using a single-metallic tube material to eliminate the need for bi-metallic welds within the tube bundle. The SG boils secondary water on the inside of the tubes and subsequently eliminates much of the phenomena responsible for SG tube degradation in current LWR U-tube designs, such as pitting corrosion on the shell side. The SG has two helium circulators mounted on the top head to provide the driving force for helium circulation, each rated for 50% of design flow.

Helium was chosen as the primary-side heat transport fluid as it is chemically and radiologically inert and has excellent heat transfer properties. It remains a single-phase gas in all conditions, thereby eliminating sudden pressure changes associated with a phase change. Unlike coolant water that undergoes density changes in the core from phase changes, helium does not affect neutron moderation since it remains in a single phase. The primary purpose of the helium and the associated helium pressure boundary defined by the Reactor Pressure Vessel, SG vessel, and helical coil tubing, circulator housings, and cross vessel, is to transport heat from the pebble bed to the SG for subsequent use in power conversion. The helium does not support a safety-related RSF to provide cooling. The reactor operates in the thermal neutron spectrum and uses graphite as the moderator. The reactor's fueled zone is comprised of a cylindrical volume filled with approximately 220,000, 60-mm diameter uranium oxycarbide (UCO) TRISO fuel pebbles. This arrangement and the core barrel graphite reflectors make up the pebble bed core, which provides capabilities for online refueling and excess reactivity management.

The Xe-100 fuel cycle is optimized by recirculating pebbles through the reactor approximately six times which enables extremely efficient fuel usage and can achieve burnup in excess of 160 GW-days/metric ton of heavy metal. The high burnup reduces the volume of spent fuel per ton of uranium and improves proliferation resistance. Pebble utilization is maximized by measuring the burnup each time a pebble passes through the reactor through gamma spectroscopy. When fuel burnup measurement indicates the pebble is nearing its burnup limit, it is sent out of the system for spent fuel storage. Spent fuel pebbles are stored in closed casks that use passive cooling to remove the spent fuel decay heat. The spent fuel pebbles require no conditioning and are in a physical state ready for long-term storage immediately upon removal from the reactor. This eliminates the need for an actively cooled spent fuel pool for storage. Fresh pebbles are added to the core each day, with an equal number being removed from the bottom of the core and placed in spent fuel casks, which can be stored on-site at the plant for the life of the plant. To better support spent fuel management at a four-unit plant, all reactor buildings are connected via a common spent fuel transfer tunnel where remotely-operated machinery moves the filled spent fuel casks to a central, mostly below-grade high-energy interim storage facility for initial cooldown in a natural draft cooled arrangement before transfer to above-ground intermediate spent fuel storage pads.

Intrinsic safety is achieved through selective design features. These design features include:

- Low excess reactivity
- A strong overall negative temperature coefficient
- Low power density

- Large thermal inertia of the core
- Passive heat removal via conduction, convection, and radiation to the environment
- A high degree of fission product retention (99.99%) during normal operation and all LBEs by the TRISO particles

Reactor decay heat removal is available through multiple passive means in the shutdown modes, ensuring long-term plant safety without the need for emergency AC/DC power.

The Xe-100 utilizes UCO TRISO-based pebble fuel, each pebble consisting of thousands of UCO TRISO-coated fuel particles embedded in matrix graphite and formed into a 60-mm diameter sphere. The fuel particle consists of a high-assay, low enriched uranium oxycarbide fuel kernel (a mixture of uranium oxide and uranium carbide phases present in a kernel) surrounded by coating layers of pyrolytic graphite and silicon carbide for retention of radionuclides. DOE initiated the AGR Fuel Development and Qualification Program in 2002, which evolved into a collaboration with the NGNP program. INL successfully completed initial irradiation campaigns (AGR 1 and 2) and, in early 2018, began final particle qualification tests in the INL Advanced Test Reactor. DOE has made over $400M of investments in the design, manufacturing, and testing of UCO TRISO fuel.

The Xe-100 safety design approach includes the design objective to limit the dose consequence for all LBEs so that regulatory dose limit targets to ensure adequate protection of public health and safety and protection of the environment are met at an Exclusion Area Boundary (EAB) that will be set 400 m from the reactor building. A further design objective is to meet the Environmental Protection Agency Protective Action Guide limits at the EAB, thereby minimizing disturbances to the day-to-day activities of nearby members of the public during any LBE. Based on previous design work for HTGRs, it is expected that meeting the Protective Action Guide limits at the EAB will likely be the limiting top-level requirement that results in setting fuel quality and performance requirements for the Xe-100 fuel.

The performance of UCO TRISO fuel sets the foundation for the Xe-100 reactor safety case. Retaining radionuclides (including fission products) in the fuel particle is critical to safe operation. Like previous HTGR designs, the fuel particles are the primary, but not the only, barrier to radionuclide release in the Xe-100. As such, X- energy recognizes the importance of the fuel qualification methodology to the design, safety case, and licensing approach. This methodology will be the subject of a licensing topical report that builds upon the 2019 topical report submitted by Electric Power Research Institute and approved by NRC in 2020 on the AGR-1/2 irradiation campaigns, as well as possible data from the AGR-5/6/7 irradiation campaign and X-energy's own experiments to confirm fuel quality and performance requirements. TRISO particles and the pebble design are shown in Figure 4**Error! Reference source not found.**.

Figure 4. TRISO Coated Particle Fuel Pebble

In contrast to LWRs, the Xe-100 does not need a pressure-retaining, essentially leak-tight containment building. It instead relies on the radionuclide retention capability provided by the fuel particles, fuel pebble, and other design features of the reactor unit. The Xe-100 radionuclide retention capabilities are schematically shown in Figure 5. These capabilities provide the technical basis and will be required to meet performance criteria to demonstrate a functional containment approach based upon fission product retention.

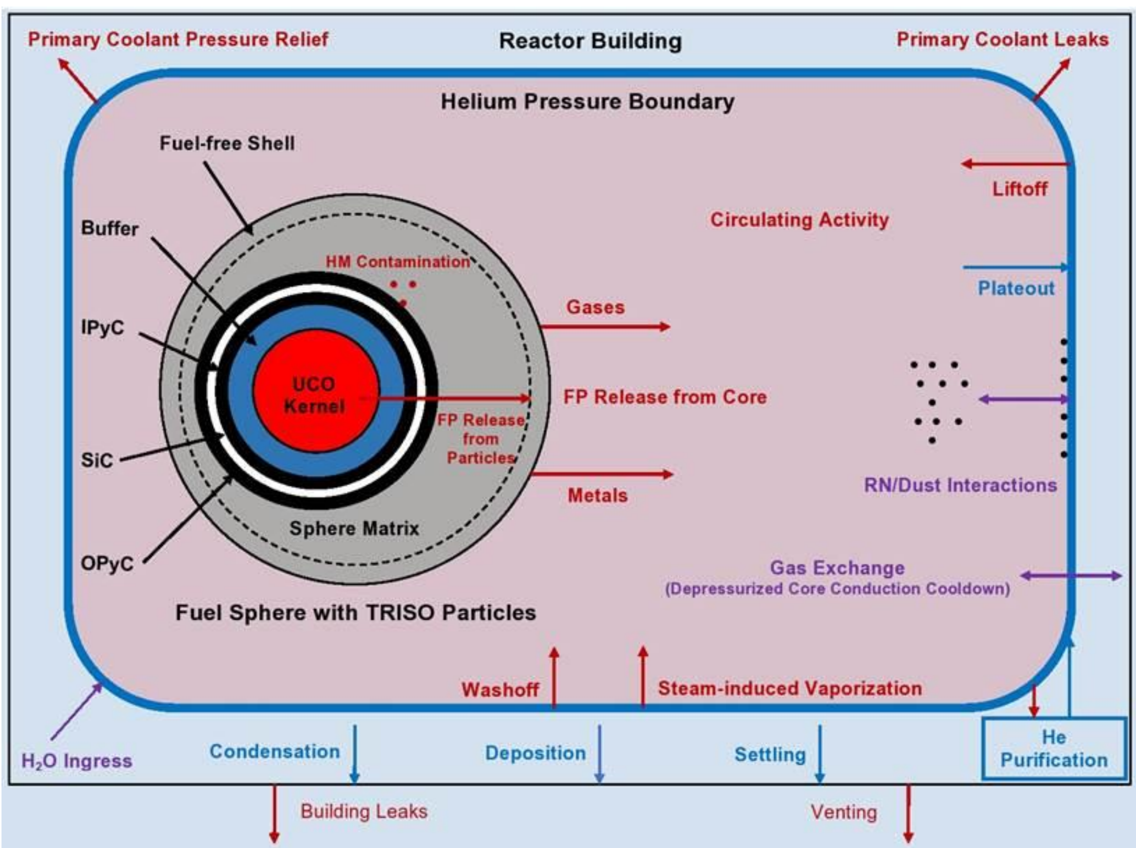

Figure 5. Xe-100 Radionuclide Retention and Release Phenomena

The five principal retention layers include: (1) the fuel kernel; (2) the particle coatings, particularly the silicon carbide coating; (3) the spherical fuel-element matrix, including the fuel-free zone; (4) the reactor helium pressure boundary; and (5) the reactor building. Each of these barriers contributes to limiting the release of radionuclides to the environment to meet the regulatory dose limit targets set as top-level requirements for the Xe-100. The contribution of each of the layers in limiting radionuclide release to the environment is calculated for each postulated LBE, depending on the response of the reactor to the event using an internally developed set of analysis codes collectively known as XSTERM. The contribution of each barrier is plant design specific and specific both to the event scenario and the radionuclide species. The various phenomena that determine radionuclide transport and release for a spectrum of postulated LBEs are also provided in Figure 5.

Collectively, the fuel pebble comprises the first three release barriers. Typically, the TRISO fuel particles retain >99.99% of the fission products even during Beyond Design Basis Events (BDBEs), which has been shown through operational experience, research and development, fuel qualification testing, as well as modeling and analyses. The release fraction of radionuclides from the particles is calculated accounting for the as-manufactured fuel quality (e.g., the allowable heavy-metal contamination and coating defects), fuel failure during irradiation, incremental fuel failure under accident conditions, and diffusion of fission products through the intact particle coatings under normal operating and accident conditions. These factors are calculated on an event-specific basis, depending on the burnup of the fuel, maximum operating temperature, the maximum temperature reached in the accident, and, where applicable, air and/or water contamination inside the helium pressure boundary (e.g., as a result of a steam generator tube rupture). The spherical fuel matrix provides additional retention of metallic fission products (e.g., retention of cesium by a factor of 10 and strontium by a factor of more than 100).

3.2 Methodologies and Analyses

The example SAR Chapter 2 content developed during the Xe-100 tabletop exercise is displayed in Appendix A of this report. A complete draft of Chapter 2 was not able to be developed given the maturity of the design and safety analyses at the time of the tabletop exercise; however, Section 4 of this report presents the major observations and feedback provided to the TICAP team during the development of the example SAR content.

The Methodologies and Analyses discussed in Appendix A (SAR Chapter 2) support the understanding of the LBEs discussed in Section 3.3 of this report and the identification of the RSFs discussed in Section 3.6 of this report.

3.3 Licensing Basis Events

The content in this section is based on the Xe-100 Phase 0 PRA model that is described in greater detail in the 2018 High Temperature Gas-Cooled Pebble Bed Reactor LMP demonstration report.[2] Based on the TICAP guidance, the LBEs would be discussed in Chapter 3 of the SAR. The following discussion of LBEs for the Xe-100 design provides context for the LBEs that are modeled using the approach outlined in Appendix A of this report. Additionally, using the RIPB approach described in NEI 18-04 and endorsed by NRC in Regulatory Guide 1.233, the results of the LBE analyses discussed in Sections 3.4 and 3.5 of this report are used to identify the RSFs discussed in Section 3.6. These RSFs are an important input to the safety classification of the SSCs, and the classification of each SSC is an important factor in which of the plant programs described in SAR Chapter 8 (Appendix B of this report) are applicable to a given SSC.

An important element of a PRA model is the systematic search for Initiating Events (IEs), which begins the process of event sequence modeling. The initial conditions for the selection of IEs for the Xe-100 PRA will eventually cover all operating and shutdown modes expected during the Xe-100 plant's operating life, including the expected shutdown configurations for conducting maintenance and inspections and for the full range of internal and external events per the ASME non-LWR standard. However, for the 2018 scope and available design information for the TICAP tabletop demonstration, IEs for the Phase 0 PRA were limited to internal events at full power. Per the ASME standard for PRA,[3] a structured Logic Diagram method is used. Table 2 lists the internal IEs identified through the Master Logic Diagram.

Table 2. Phase 0 PRA Internal Initiating Events at Full Power

Internal Initiating Events
Turbine Trip (TT)
Reactor Trip (RT)
Circulator Trip (CT)
Loss of Primary Flow (LF)
Control Rod Withdrawal (CR)
Loss of Offsite Power (LO)
Steam Generator Feedwater Pump Trip (FW)
Small Helium Depressurization (SD)
Medium Helium Depressurization (MD)
Large Helium Depressurization (LD)
Steam Generator Tube Rupture (SG)

Event sequence diagrams and event trees are constructed for each IE category. The Small Helium Depressurization Event Tree, shown in Figure 6**Error! Reference source not found.**, illustrates a typical event tree from the Phase 0 PRA with the IE on the left (in units of per-plant-year for the four-reactor module plant) and each of the branch points sequentially across the top for the plant response. For each branch point question, the Yes-No branches are shown with their estimated probability (no units). The final columns provide the overall event sequence frequency and the associated risk-informed LBEs in the three frequency ranges.

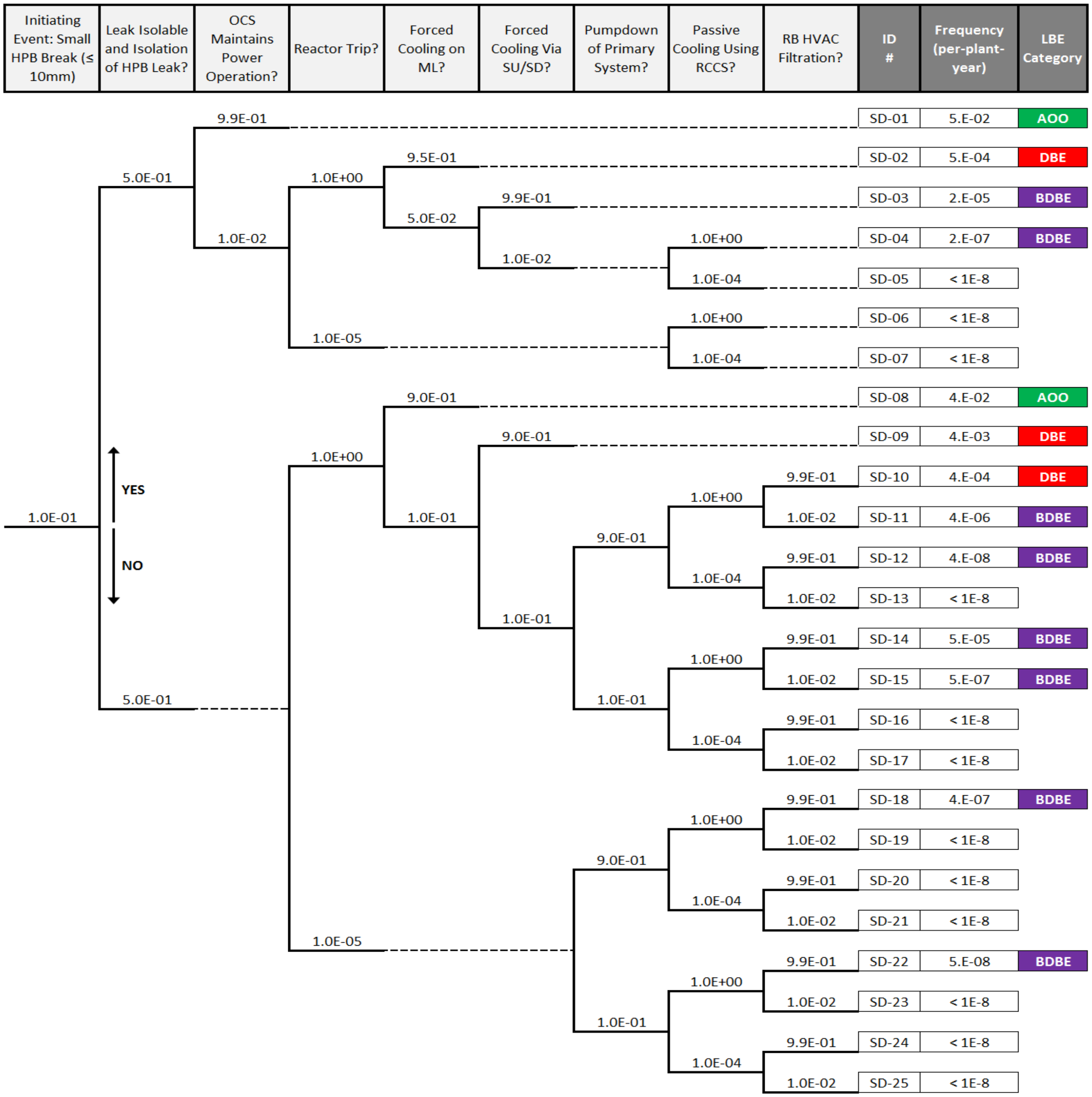

Figure 6. Small Helium Depressurization Event Tree with Associated LBEs

For the IEs in Table 2, a total of 11 Anticipated Operational Occurrences (AOOs), 17 Design Basis Events (DBEs), and 34 BDBEs were identified in the Phase 0 PRA. Of those, two AOOs, five DBEs, and 23 BDBEs result in a release of radionuclides.

The IEs, including small, medium, and large primary depressurization and steam generator feedwater trip, are those which involve a Helium Pressure Boundary (HPB) challenge. If the HPB is breached, a radionuclide release into the Reactor Building (RB) with the potential to be released to the environment occurs due to differential pressure across the system. All other event sequences with an intact HPB do not lead to a release of radionuclides. For the Xe-100, the releases of radionuclides are categorized, in order of ascending magnitude, as:

- Initial release from the HPB of circulating activity in the primary helium: All sequences initiated by HPB leaks and breaks involve at least some or all of the circulating activity.
- Initial release of normal operation plateout activity from the primary internal surfaces for larger breaks that involve shear forces during depressurization.
- Delayed release from pebble fuel activity: If the sequence involves a loss of forced circulation, for example, a release delayed on the order of hours to tens of hours may occur during the increase of core temperatures during the passive heat removal from the reactor vessel based upon fuel performance during the event sequence.
- Increased release from events with steam/water ingress events into the reactor: For sequences involving a steam generator tube rupture with steam/water ingress to the primary system with subsequent pressure relief to the reactor building, some of the circulating activity would be released and plateout may be washed off of primary internal surfaces and available for release.

3.4 Consequences for the LBEs Identified in the Phase 0 PRA

As the design and safety analysis of the Xe-100 matures, the analysis methods documented in Appendix A of this report (SAR Chapter 2) will be used to calculate specific consequences for each of the LBEs identified above; however, at the time of this tabletop exercise, the most recent Xe-100 consequence estimates associated with the Phase 0 PRA were developed in a less design-specific manner to enable the performance of those tasks in the RIPB process that utilize quantitative frequency-consequence estimates before the more accurate consequence values are available. According to the TICAP guidance, the consequences of each LBE would be summarized in Chapter 3 of the SAR, and the integrated evaluations of plant risk would be presented in Chapter 4 of the SAR.

Table 3 lists the LBEs that were identified in the Phase 0 PRA, a brief description of the event sequences, their estimated frequency, and scaled dose at the Exclusion Area Boundary using either the MHTGR (M) or PBMR (P) consequence data to provide the basis for the estimate. As shown, there are 11 AOOs, 17 DBEs, and 34 BDBEs. In order to produce these consequence estimates, mechanistic off-site consequence data from prior HTGR PRAs were scaled by core thermal power level. Consequence values from the 350 MWt prismatic MHTGR were scaled by a factor of 0.57 and from the 268 MWt pebble PBMR by a factor of 0.75. The design and PRA mature event sequences will be grouped based on similarity of plant response, end state, and magnitude of the mechanistic source term.

Table 3. LBEs for the Xe-100 Grouped by Event Sequence Frequency

LBE	PRA ID	LBE Description	Event Sequence Frequency, per plant-yr	Dose, WB rem	Dose Basis, PBMR/ MHTGR
Anticipated Operational Occurrences					
1	TT-01	Turbine trip, plant runback to reduced power level	1×10^{1}	$< 10^{-5}$	--
2	RT-01	Reactor trip, forced cooling via main-loop system	6×10^{0}	$< 10^{-5}$	--
3	CT-01	Circ. trip, forced cooling via main-loop system	4×10^{0}	$< 10^{-5}$	--
4	CT-02	Circ. trip, forced cooling via start-up/shutdown (SU/SD) system	4×10^{-1}	$< 10^{-5}$	--
5	RT-02	Reactor trip, forced cooling via SU/SD system	3×10^{-1}	$< 10^{-5}$	--
6	LO-01	Loss of Offsite Power, plant maintains house load	1×10^{-1}	$< 10^{-5}$	--
7	TT-02	Turbine trip, forced cooling via main-loop system	9×10^{-2}	$< 10^{-5}$	--
8	FW-01	Feedwater (FW) trip, forced cooling via SU/SD system	5×10^{-2}	$< 10^{-5}$	--
9	SD-01	Sm. Helium Leak, isolated, plant maintains operation	5×10^{-2}	1×10^{-5}	P
10	SD-08	Sm. Helium Leak, no isolation, forced cooling via main loop	5×10^{-2}	1×10^{-5}	P
11	CT-03	Circ. trip, forced cooling failure, passive cooling via Reactor Cavity Cooling System (RCCS)	2×10^{-2}	$< 10^{-5}$	--
Design Basis Events					
1	SG-01	SG tube rupture, leak isolation, forced cooling via SU/SD system	9×10^{-3}	1×10^{-5}	--
2	CR-01	Rod withdrawal, forced cooling via main loop	9×10^{-3}	$< 10^{-}5$	--
3	LO-02	Loss of Offsite Power < 3 hr, forced cooling via SU/SD	5×10^{-3}	$< 10^{-5}$	--
4	TT-03	Turbine trip, forced cooling via SU/SD system	5×10^{-3}	$< 10^{-5}$	--
5	SD-09	Sm. Helium Leak, no isolation, forced cooling via SU/SD system	5×10^{-3}	1×10^{-4}	--
6	RT-03	Reactor trip, passive cooling via RCCS	3×10^{-3}	$< 10^{-5}$	--
7	FW-02	FW trip, passive cooling via RCCS	5×10^{-4}	$< 10^{-5}$	--
8	CR-02	Rod withdrawal, forced cooling via SU/SD	5×10^{-4}	$< 10^{-5}$	--
9	MD-01	Medium Helium Break, isolation, forced cooling via SU/SD	5×10^{-4}	3×10^{-5}	P
10	SD-02	Small Helium Leak, isolation, forced cooling via main loop	5×10^{-4}	1×10^{-5}	P
11	SD-10	Small Helium Leak, passive cooling via RCCS, primary pump down successful	5×10^{-4}	2×10^{-4}	M
12	MD-02	Medium Helium Break, no isolation, forced cooling via SU/SD	5×10^{-4}	3×10^{-5}	P
13	LO-09	Loss of Offsite Power < 24 hr., forced cooling via SU/SD	4×10^{-4}	$< 10^{-5}$	--
14	LF-01	Loss of Offsite Power, passive cooling via RCCS	4×10^{-4}	$< 10^{-5}$	--
15	LO-05	Loss of Offsite Power < 3 hr., passive cooling via RCCS	4×10^{-4}	$< 10^{-5}$	--
16	LO-03	Loss of Offsite Power < 3 hr., passive cooling via RCCS	3×10^{-4}	$< 10^{-5}$	--
17	LO-16	Loss of Offsite Power > 24 hr., forced cooling via SU/SD	2×10^{-4}	$< 10^{-5}$	--
Beyond Design Basis Events					
1	SG-02	SG tube rupture, leak isolation and dump, forced cooling via SU/SD	9×10^{-5}	1×10^{-5}	M
2	SG-04	SG tube rupture, isolation, dump stuck open, forced cooling via SU/SD	9×10^{-5}	1×10^{-5}	M
3	SG-18	SG (tube rupture), no isolation, forced cooling via SU/SD	9×10^{-5}	2×10^{-4}	M

LBE	PRA ID	LBE Description	Event Sequence Frequency, per plant-yr	Dose, WB rem	Dose Basis, PBMR/ MHTGR
4	SG-09	SG (tube rupture), isolation, dump fails to open, forced cooling via SU/SD	9×10^{-5}	1×10^{-4}	M
5	SD-14	Small Helium Leak, no isolation, passive cooling via RCCS	5×10^{-5}	4×10^{-4}	M
6	TT-04	Turbine trip, passive cooling via RCCS	5×10^{-5}	$< 10^{-5}$	--
7	MD-14	Medium Helium Break, passive cooling via RCCS	5×10^{-5}	2×10^{-4}	M
8	FW-04	FW trip, circ. trip fails, passive cooling via RCCS	4×10^{-5}	$< 10^{-5}$	--
9	CT-05	Circ. trip, passive cooling via RCCS	4×10^{-5}	$< 10^{-5}$	--
10	LO-12	Loss of Offsite Power < 24 hr., passive cooling via RCCS	3×10^{-5}	$< 10^{-5}$	--
11	SD-03	Small Helium Leak, isolation, forced cooling via SU/SD	3×10^{-5}	1×10^{-5}	P
12	LO-10	Loss of Offsite Power < 24 hr., passive cooling via RCCS	2×10^{-5}	$< 10^{-5}$	--
13	LO-19	Loss of Offsite Power > 24 hr., passive cooling via RCCS	2×10^{-5}	$< 10^{-5}$	--
14	FW-06	FW trip, circ. trip fails, passive cooling via RCCS	1×10^{-5}	$< 10^{-5}$	--
15	SG-20	SG tube rupture, no isolation, open Pressurizer Safety Relief Valve (PSRV), passive cooling via RCCS	1×10^{-5}	2×10^{-2}	M
16	SG-12	SG tube rupture, isolation, stuck open PSRV, forced cooling via SU/SD	1×10^{-5}	5×10^{-3}	M
17	LO-17	Loss of Offsite Power > 24 hr., passive cooling via RCCS	9×10^{-6}	$< 10^{-5}$	--
18	FW-12	FW trip, circ. trip fails, open PSRV, passive cooling via RCCS	7×10^{-6}	$< 10^{-5}$	--
19	LD-01	Large Helium Break, passive cooling via RCCS, RB success	7×10^{-6}	8×10^{-5}	M
20	MD-02	Medium Break, isolation, forced cooling via SU/SD	6×10^{-6}	3×10^{-5}	M
21	SD-11	Small Helium Leak, no isolation, passive cooling via RCCS, pump down	6×10^{-6}	2×10^{-4}	P
22	MD-12	Medium Helium Break, no isolation, passive cooling via RCCS	6×10^{-6}	3×10^{-4}	P
23	CR-03	Rod withdrawal, passive cooling via RCCS	5×10^{-6}	$< 10^{-5}$	--
24	MD-26	Medium Helium Break, no isolation, forced cooling via SU/SD	5×10^{-6}	2×10^{-4}	M
25	MD-03	Medium Helium Break, isolation, passive cooling via RCCS	5×10^{-6}	3×10^{-5}	M
26	CT-04	Circ. trip, passive cooling via RB	2×10^{-6}	$< 10^{-5}$	--
27	LD-02	Large Helium Break, passive cooling via RCCS, RB failure	2×10^{-6}	8×10^{-5}	M
28	LD-09	Large Helium Break, passive cooling via RCCS, RB failure	1×10^{-6}	8×10^{-5}	M
29	SG-05	SG tube rupture, isolation, dump stuck open, passive cooling via RCCS	1×10^{-6}	2×10^{-5}	M
30	TT-06	Turbine trip, passive cooling via RCCS	1×10^{-6}	$< 10^{-5}$	--
31	SG-10	SG tube rupture, leak isolation, open PSRV, forced cooling via SU/SD	1×10^{-6}	2×10^{-5}	M
32	SG-25	SG tube rupture, no isolation, no FW trip, forced cooling	8×10^{-7}	8×10^{-4}	M
33	SD-15	Small Helium Leak, no isolation, passive cooling via RCCS	6×10^{-7}	4×10^{-4}	M
34	MD-15	Medium Helium Break, no isolation, passive cooling via RCCS	6×10^{-7}	2×10^{-4}	P

3.5 Integrated Evaluation with the LMP Risk Target

Using the LBEs and the frequencies and allocated consequence values from Table 3, Figure 7 provides the comparison of the Xe-100 risk with the proposed Frequency-Consequence Target (F-C Target). According to the TICAP guidance, integrated evaluations of the LBE frequencies and consequences, as well as evaluation of the adequacy of plant and programmatic DID, would be discussed in Chapter 4 of the SAR.

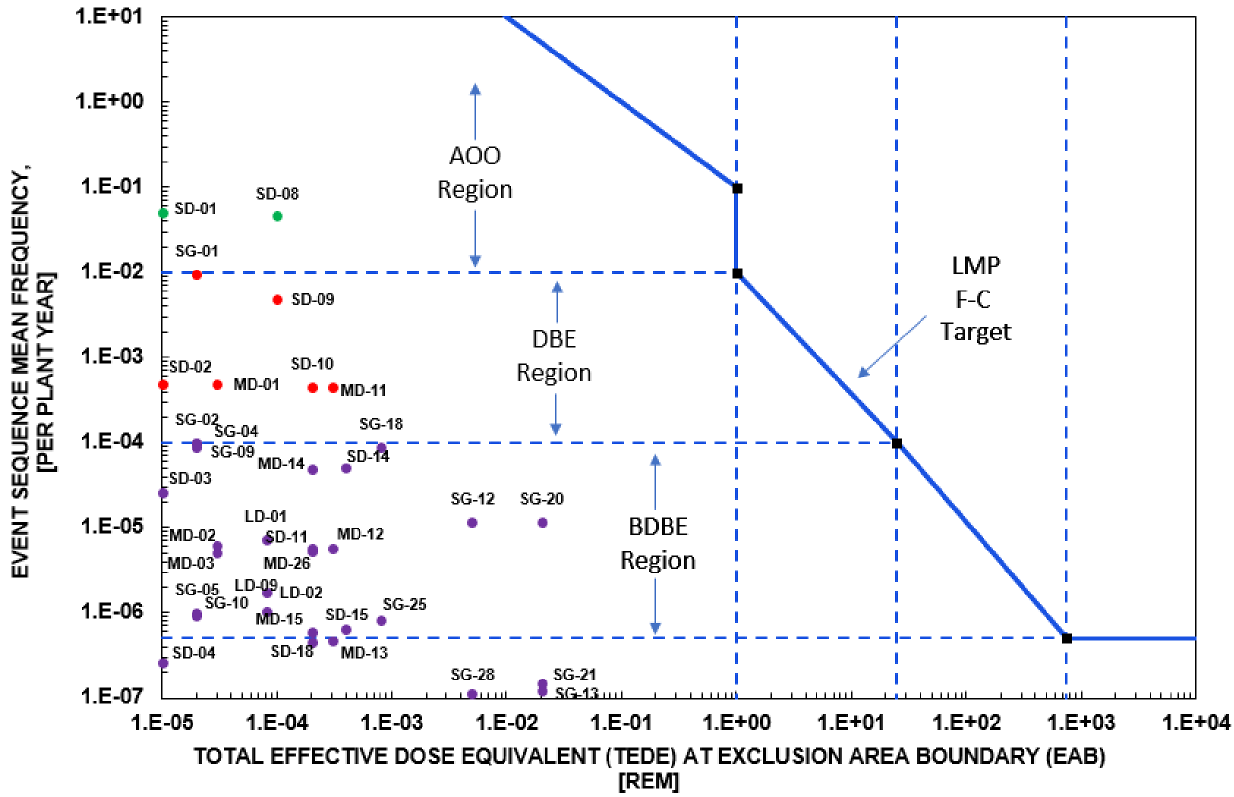

Figure 7. Xe-100 Risk Comparison to the LMP F-C Target

As illustrated in Figure 7, there are large margins between the LBE expected value points and the target line. The expected value points are three to four orders of magnitude apart in risk, and the highest consequence LBEs are two orders of magnitude from the design goal—namely, the 1-rem Protective Action Guide criterion for sheltering and/or evacuation of the off-site public.

There was sufficient breadth in the Phase 0 PRA, even at the early stage of the Xe-100 design, to identify a spectrum of event sequences in each of the AOO, DBE, and BDBE categories with varying releases in terms of sources, pathways, magnitude, and timing. These LBEs and their associated event sequences provide the basis for the necessary types of methodologies and analyses required to be described in Chapter 2 of the TICAP-informed SAR.

3.6 Derivation of Required Safety Functions

This section derives the RSFs from the prior selection of LBEs, based upon PRA safety functions, as a necessary step in the development of the RFDC and the safety classification of plant SSCs. According to the TICAP guidance, both the design-specific RSFs and RFDC are presented in Chapter 5 of the SAR. The example SAR content in Appendix B of this report (Chapter 8 of the SAR) is intended to describe some of the programs that are used for special treatments for Safety-Related (SR) or Non-Safety-Related with Special Treatment (NSRST) SSCs (described in SAR Chapters 6 and 7, respectively).

In general, the safety functions of SSCs are responsible for preventing and mitigating the release of radioactive material from any radionuclide source. The general safety functions are hierarchical, with the highest level applicable to nuclear power plants in general and the bottom level applicable to the Xe-100.

Figure 8 includes safety functions for the protection of the personnel within the plant and the public off-site developed during the conceptual design phase. It also includes radiation sources not only in the core but also within the plant processes and in onsite storage. The multiple barriers for protection of the off-site public from the Xe-100 reactor core are shown in the middle of the chart. At the Xe-100 lower level, the required safety functions are expanded within the core, where the radionuclide retention is by design focused within the ceramic-coated fuel particles with the three key safety design functions highlighted at the bottom of the figure: Control Heat Removal, Control Core Criticality,* and Control Water/Steam Ingress.

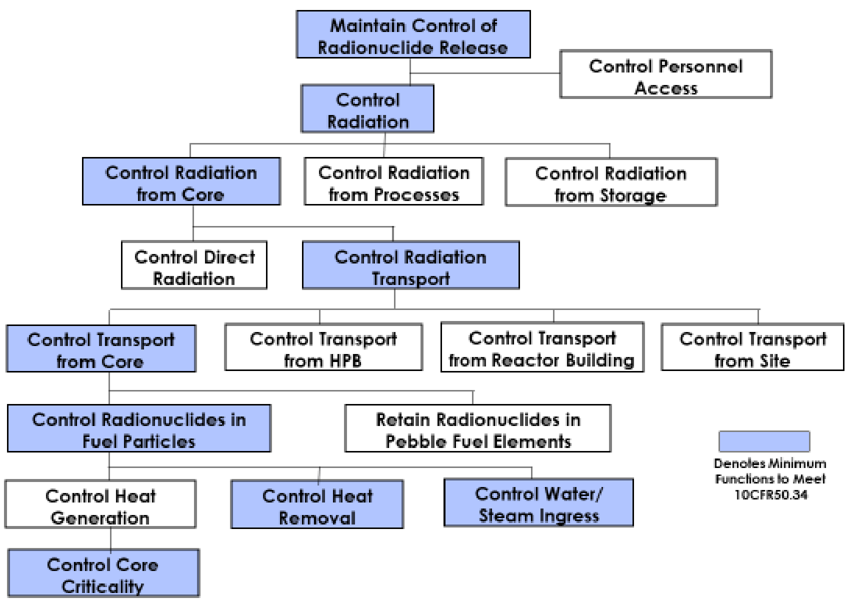

Figure 8. Conceptual Design Xe-100 General and Required Safety Functions

3.7 SSC Safety Classification and Design Details

According to the TICAP guidance, Chapter 6 of the SAR provides further detail on the criteria and capabilities of all SR SSCs in the LMP-based affirmative safety case, consistent with the NEI 18-04 methodology, and Chapter 7 of the SAR does the same for each NSRST SSC.

* At the time of the TICAP tabletop, an active integrated decision making process team discussion was on-going to consider "Control Reactivity" as an alternative RSF to "Control Core Criticality"

Although the example SAR content presented in Appendix B of this report (SAR Chapter 8) is intended to describe the plant programs that are relied upon as special treatments for SR and/or NSRST SSCs, the example content in this tabletop exercise is sufficiently general such that it is not necessary to provide context regarding any specific SSCs or its safety classification.

3.8 Plant Programs

The example SAR content developed as part of the Xe-100 tabletop exercise for Chapter 8 of the SAR is contained in Appendix B of this report. According to the TICAP guidance, the content of SAR Chapter 8 is intended to describe the programs relied upon to provide reasonable assurance that (i) reliability and performance targets are met throughout the plant lifetime and (ii) safety-significant uncertainties are effectively addressed as part of DID. Although SAR Chapter 8 should include a discussion of such programs as applied across different plant lifetime phases (i.e., design, construction, testing, and operations), only programs pertaining to the design stage of the Xe-100 were within the scope of this tabletop exercise. The major observations and lessons learned from the development of the example SAR content are presented in Section 4 of this report.

4.0 OBSERVATIONS AND CONCLUSIONS

4.1 Observations and Lessons Learned

PRA Content for the SAR

The Xe-100 tabletop elicited good dialogue on the scope, level of detail, and types of information that would be provided related to the design PRA in the SAR. Since the Xe-100 was in an early phase of design, the maturity of the PRA and associated risk insights were commensurate. Risk-insights are (or would be) provided in several locations throughout the SAR, and so the content requirements in Chapter 2 could be reduced. Additionally, the extensive documentation on the PRA would remain available for audit and review outside of the SAR, managed via the processes described in the non-LWR PRA standard.

Simplification of Complex Processes

Some content drawn into the example Chapter 2 was initially drawn from a topical report under development. This topical report described the Xe-100 source term development methodology and included an extensive flow chart of all the various phenomena of interest that the X-energy code suite can model. While technically interesting and complete, during the tabletop review, it became apparent that it was too much detail and added confusion to the description of significant phenomena in the plant. A pointer to the figure in the associated topical report or a simplified graphic may have been more appropriate. The nature of the TICAP structure and content guidance opens up value-added opportunities to ensure SARs are written with concision and completeness.

Safety Analysis Details

The descriptions of many general analysis methodologies and approaches were developed at a preliminary level of detail during the tabletop exercise, but it was recognized that Regulatory Guide 1.203 elements would be best placed in Chapter 2, including phenomena uncertainties, evaluation model development, and sensitivity analyses and risk-significance discussion. Some of this content could be placed in subsequent chapters (i.e., LBE evaluations) as an alternative.

Quantitative Data

The description of mechanistic source term development has the potential to include a significant amount of quantitative data in the form of release fractions, uncertainties, leak path factors, etc. The team dialogue around whether to place such detailed content in the SAR, and the necessary context for a reader to understand its basis, should be weighed against the placement of such details in technical or topical reports that naturally provide such context.

Lastly, the Xe-100 tabletop generated a good dialogue concerning the inclusion of reliability, capability, and availability information in various parts of the SAR, and that Chapter 8 provides a logical place to align programmatic special treatments with that information. One such programmatic activity is the development of Technical Specifications as a means of protecting reliability and capability targets. The necessary guidance to develop LMP-based Technical Specifications was outside the scope of TICAP but was initiated by the NRC staff in May 2021.

4.2 Conclusions

X-energy's participation in the TICAP tabletop exercise activity was a valuable experience for the licensing and analysis teams. It was a practical opportunity to work through LMP-based content development and decision-making. While lesser in scope than anticipated, it produced valuable outcomes early in the tabletop series that added value for all participants and the prospective industry user base. 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

[1] Nuclear Energy Institute, "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development," NEI 18-04, Revision N, 2018.

[2] Southern Company, "High Temperature, Gas-Cooled Pebble Bed Reactor Licensing Modernization Project Demonstration," SC-29980-200, ADAMS Accession Number ML18228A779, August 2018.

[3] ASME/ANS Joint Committee on Nuclear Risk, "Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants," ASME/ANS RA-S-1.4-2020, 2020.

[4] Idaho National Laboratory, "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, Probabilistic Risk Assessment Approach," Rev 0, August 2019.

[5] Idaho National Laboratory, "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, Selection and Evaluation of Licensing Basis Events," Rev 0, August 2019.

[6] Idaho National Laboratory, "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, Safety Classification and Performance Criteria for Structures, Systems and Components," Rev 0, August 2019.

[7] Idaho National Laboratory, "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, Risk-Informed and Performance-Based Evaluation of Defense-in-Depth Adequacy," Rev 0, August 2019.

[8] Nuclear Energy Institute, "Technology Inclusive Guidance for Non-Light Water Reactor Safety Analysis Report: Content for a Licensing Modernization Project-Based Affirmative Safety Case," Rev 0, To be published September 2021.

Appendix A Draft Content for SAR Chapter 2 – Generic Analyses

This chapter describes the generic analyses that support the safety design approach for the Xe-100 reactor. These analysis methods, approaches, and tools were used to identify licensing basis events, evaluate the consequences of such events, evaluate the performance of safety-related and NSRST Structures, Systems, and Components (SSCs), and assess the integrated plant's response to normal and off-normal operating conditions. The descriptions of these analysis methods are cross-cutting through the entire SAR and support multiple LBEs and SSCs. Licensing topical reports are identified for several analysis methodologies, and approaches, and their incorporation by reference is described herein.

A.1. Probabilistic Risk Assessment

X-energy's Probabilistic Risk Assessment (PRA) was developed in accordance with the ASME-ANS Advanced Non-Light Water Reactor PRA Standard [ASME/ANS RA-S-1.4-2020] and matured through Capability Category 2 for use in the licensing bases and risk-informed applications. This standard specifies requirements for having a technically adequate PRA for all stages of design, construction, and operation. The basis for this standard comes from many existing LWR PRA standards and principles that were transformed to be technology neutral. Notably and prior to this standard, PRA has been used successfully with HTGRs such as the MHTGR-350, PBMR, and SC-HTGR. Valuable insights and lessons learned from those projects have influenced the PRA development for the Xe-100, which is summarized here. Significant insights include:

- Identification of initiating event categories (What can go wrong?)
- Identification of License Basis Events (LBEs) and their respective frequencies (How likely is it?)
- Identification of distinct end-states and source term categories (What are the consequences?)

A.1.1 Overview of PRA

In support of the Combined License (COL) application, the Xe-100 PRA was developed to analyze full power conditions, the modes/states associated with hot standby conditions, and some cold shutdown conditions when identified as risk-significant contributors to off-site dose. All significant sources of radionuclides were evaluated using screening criteria for further analysis of initiating events and event sequences.

[Further discussion of implementation of ANLWR standards….]

The Xe-100 PRA is documented in a set of technical reports, principally "Xe-100 Probabilistic Risk Assessment: Final Design Phase" [XE00-] and associated software models. This report is a general reference to the safety analysis report.

A.1.2 Conformance to ANLWR PRA Standard and Regulatory Guidance

The Xe-100 was developed to conform to all elements of the ASME/ANS ANLWR PRA Standard, 2020 edition, with the following clarifications:

[Discussion of clarifications. At the time of the TICAP tabletop, the NRC staff's endorsement of the standard was ongoing, and no clarifications were developed.]

The NRC endorsed the ANLWR PRA Standard through Regulatory Guide (RG) 1.2xx with clarifications. The Xe-100 PRA incorporates all clarifications in the following manner:

[Discussion of such clarifications or exceptions. At the time of the TICAP tabletop, the NRC staff's endorsement of the standard was ongoing, and no clarifications were developed.]

A.1.3 PRA Peer Review

The Xe-100 PRA used for the COL application was peer reviewed in accordance with NEI 20-09 by a team of experienced practitioners in PRA, HTGR technology, and other areas of expertise. The peer review was conducted during a series of meetings in [Date, year] and documented in the latest Xe-100 PRA report. Specific feedback from the peer review was incorporated into Revision [xx] by [Date, year] for application in the COL development effort.

A.1.4 PRA Scope

[No content developed for this subsection as part of this exercise]

A.1.4.1 Radionuclide Sources

[Input]

[Internal initiating event sequences consider all RN sources in the reactor system, including those in the fuel, circulating activity, and activation products. Spent fuel, fresh fuel, radioactive waste are considered…]

A.1.4.2 Multi-Unit Considerations

[No content developed for this subsection as part of this exercise]

[Either X-energy report or ANLWR PRA standard]

A.1.4.3 Internal Hazard Screening

[No content developed for this subsection as part of this exercise]

A.1.4.4 External Hazard Screening

[No content developed for this subsection as part of this exercise]

A.1.4.5 Plant Operating States

[No content developed for this subsection as part of this exercise. Descriptions of expected safe endstates and stable conditions would be provided]

A.1.5 PRA Analysis Tools

The Xe-100 PRA is primarily managed in the RISKMAN software package for all plant operating states. Additional analysis for comparison of results and sensitivity was conducted using the SAPHIRE code suite. Details of each code's implementation and quality assurance are provided in the Xe-100 PRA Report [XE00-xxxxxxxx]. Source terms and radiological consequences were developed using the methodology described in Section 2.2.

A.1.6 Summary of Key PRA Results

Because NEI 18-04 is a risk-informed methodology, key PRA results are incorporated in the descriptions of the outputs of the methodology provided later in the SAR. Those results are not repeated here, but this section provides pointers to those PRA results.

- Chapter 3 presents LBEs that are supported by event sequences in the PRA. It includes 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.
- Chapter 4 shows the integrated risks across all of the LBEs and compares them to the NEI 18-04 cumulative risk metrics. It also describes the DID evaluation, which is informed by uncertainties in the PRA results.

A.2. Mechanistic Source Term Development

The Xe-100 approach to developing mechanistic source terms is provided in detail in the topical report "Xe-100 Mechanistic Source Term and Functional Containment Approach" [XE00-R-R1ZZ-RDZZ-L-000632], which is incorporated by reference in this application. A summary of the approach and major phenomena are described herein. The approach is organized into two parts. The first focuses on fuel performance and radionuclide transport in fuel particles and fuel pebbles up to the release of radionuclides from the surface of the fuel pebble into the surrounding gas space. The second focuses on radionuclide transport phenomena outside of the fuel in the helium pressure boundary, reactor building, and the environment surrounding the outside of the reactor building. The approach has been structured to be generally applicable for any gas-cooled reactor in that it highlights important phenomena and is based on similar approaches taken by predecessor HTGRs like MHTGR, PBMR, and NGNP.

Figure A-1 is a flow chart accounting for all major sources of radionuclide release from the fuel element. Classes of events that initiate radionuclide release from TRISO fuels were identified as follows: as-fabricated defects, normal in-service releases, inert accidents (e.g., core-conduction cooldown events), and oxidation accidents (i.e., air-ingress and moisture-ingress). Each class of initiator was further subdivided into specific fuel characteristics (as-fabricated SiC defects, as-fabricated uranium contamination, etc.), fuel performance phenomena (in-service and inert accident SiC failure, TRISO failure, etc.), and transport phenomena (e.g., diffusion). Next, sources of relevant data were identified and linked to fuel performance and transport phenomena, and these phenomena were linked so as to result in a final value for fission product release at the end of the analysis for a particular release initiator. This value serves as input to the ex-fuel transport models discussed in the MST topical report.

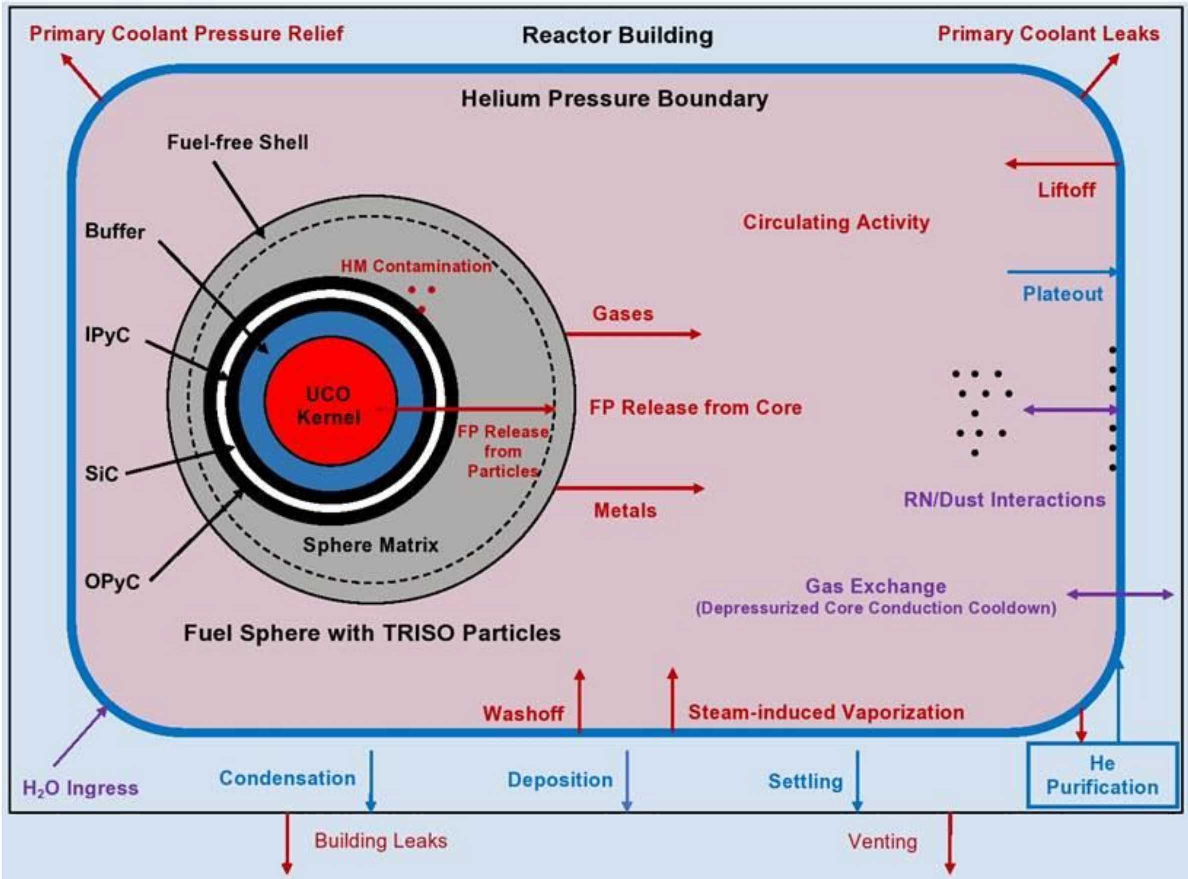

Figure A-1. Pebble-Bed HTGR Radionuclide Retention System

A.2.1 Included Transport and Other Phenomena

For nominal, steady-state operation, an R/B model for noble gases (i.e., isotopes of Kr and Xe) and iodine is employed. For transient events (i.e., accidents), a diffusion model for these elements is employed. A kernel diffusion model is employed for cesium, europium, silver (for which other models also exist), and strontium for all release-initiating events except for dispersed uranium. Fission products from dispersed uranium are produced outside of TRISO coatings and thus bypass kernel, PyC, and SiC models. A PyC diffusion model is employed for cesium, europium, and strontium for cases of as-fabricated SiC defects, in-service intact particle releases, in-service SiC failures, and SiC failures from inert/oxidation accidents. A SiC layer diffusion model is employed for cesium, europium, and strontium for in-service intact particle releases. Matrix diffusion is modeled for cesium, europium, and strontium in all cases. Inventories of fission products retained in the kernel, TRISO layers, and matrix during normal operation may be reevaluated in the case of an inert or oxidation accident. Corrosion models that account for fission product releases due to kernel oxidation and matrix oxidation are also included.

A.2.2 As-fabricated Releases

At the start of irradiation, as-fabricated fuel has three characteristics that may contribute to fission product release: SiC defects, exposed kernels, and distributed uranium (also known as "dispersed" uranium, uranium contamination, or heavy metal contamination). The frequency of SiC defect and exposed kernel occurrence is given as a fraction of the total number of fuel particles within a fuel element. The amount of dispersed uranium is given as a fraction of the total amount of uranium in a fuel element. These values are determined in the course of fuel characterization and quality control. Prior to fuel fabrication, acceptable values for each of these defects are typically expressed in the fuel fabrication specification. Either the fuel characterization or specification could be used to provide these fractions. Typical SiC defects, exposed kernels, and dispersed uranium (matrix contamination) fractions are 5E-5, 5E-5, and 1E-5, respectively.

[Description of Fuel Qualification topical report content on as-fabricated releases]

A.2.3 In-service Releases

There are three modes of fission product release during normal operations: release through intact TRISO coatings, releases due to SiC failures, and releases due to TRISO failures. In-service SiC and TRISO failures are different from as-fabricated SiC defects and exposed kernel defects, respectively, because they are failures of layers that were initially intact at the time of fabrication. AGR PIE routinely quantifies the fraction of particles with SiC layers and TRISO coatings that failed during irradiation. Combining all of the AGR-1 and AGR-2 PIE data at 95% confidence gives an in-service SiC failure fraction of about…[additional details would be provided commensurate with the COL level of analysis maturity].

For intact particle releases, the third mode of fission product release from normal, in-service conditions, it has been empirically shown that noble gases and iodine are not released; however, europium, strontium, and silver may be released. Europium and strontium releases occur particularly at higher irradiation temperatures (>1,100°C), but significant quantities of silver may be released at lower irradiation temperatures. Transport calculations validated with PIE data are used to assess releases of these elements from in-service, intact particles.

A.2.4 Inert Environment Event Sequences

Inert accidents are cases where high primary system temperatures occur, but the fuel remains under a helium atmosphere. The majority of particles may remain intact, and diffusion of fission products through these intact particles at accident temperatures (not depicted in the flow chart) could be calculated. Inventories of fission products retained in the kernel, TRISO layers, and matrix during normal operation may be reevaluated at the high temperatures of an inert accident. Some fraction of particles may have SiC or TRISO failures. Extensive AGR post-irradiation "safety" testing at temperatures of 1,600 to 1,800°C has provided statistical probabilities of SiC and TRISO failures. Combining the 1,600°C safety test results from AGR-1 and AGR-2 at 95% confidence gives SiC failure rates of approximately 1.7E-4 and TRISO failure rates of approximately 6.6E-5. Additional margin was applied to these values so that the SiC failure rate at 1,600°C is given as 3E-4, and the TRISO failure rate is given as 8E-5. Use of these bounding empirical values is used in several cases in lieu of a mechanistic fuel performance model. In certain cases, the XSTERM module XFP is used, and SiC and TRISO failure fractions could be calculated.

A.2.5 Oxidation Event Sequences

Oxidation accidents include air-and moisture-ingress events, and when analysis shows these to be design-basis events, then models for the resultant oxidation phenomena are used in the safety analyses. Alternatively, bounding assumptions may be appropriate in place of phenomenological models. An air-ingress event may be initiated by a break in a primary system coolant pipe (or multiple pipes) followed by system depressurization and air infiltration. A moisture-ingress event may be initiated by a break in a steam generator tube. Air and moisture may cause fuel oxidation. These events may also involve temperatures above normal operation. Some particles may remain intact, in which case, diffusion of fission products through these intact particles at accident temperatures (not depicted in the flow chart) could be calculated. Additionally, inventories of fission products retained in the kernel, TRISO layers, and matrix during normal operation may be reevaluated at the high temperatures of the accident. Additional release due to fuel oxidation is accounted for via modeling in XSTERM and/or bounding assumptions based on experimental data. For example, PIE has shown that under normal irradiation conditions, the matrix may retain significant quantities of silver, europium, and strontium. The releases of these matrix inventories may be in some way related to the extent of matrix oxidation, and a model of that phenomenon is included. (Historically, only fission gas release due to fuel oxidation has been measured. Condensable fission product releases from fuel oxidation have not been measured.) Experimental results are used to bound SiC and TRISO failure rates for air-and moisture-ingress events in lieu of a mechanistic model of fuel failure/degradation due to oxidation.

A.2.6 Simplified Empirical Approach

High-quality TRISO fuel is known for its retention of both condensable and gaseous radionuclides under both normal operation and accident conditions with temperatures well in excess of normal operating temperatures [65]. A conservative approach to generating a source term and/or judging the conservatism of a calculated source term would be to assemble bounding experimental results of fission product releases from all relevant sources, take no credit for radionuclide holdup outside of the fuel, and compare this to regulatory limits and reactor designer goals.

For as-fabricated SiC defects, in-service SiC failures, and inert accident SiC failures, 100% release of condensable fission products is assumed, but no release of noble gases and iodine is assumed because at least one PyC layer remains intact. For as-fabricated exposed kernels, as-fabricated dispersed uranium, in-service TRISO failures, and inert accident TRISO failures, 100% release of condensable fission products, noble gases, and iodine is assumed. The rates of these defects and failures are taken from Figure 1 of [XE00-R-R1ZZ-RDZZ-L-000632]. For normal, in-service releases from intact fuel, the fuel compact release fractions averaged over AGR-2 Capsules 5 and 6 were used. Capsule 5 fuel compacts had a TAVA irradiation temperature of 1,101°C, and Capsule 6 compacts had a TAVA irradiation temperature of 1,074°C. These temperatures are significantly higher than the average Xe-100 fuel temperatures, making this a conservative approach. These values are compared to regulatory limits and design goals for Xe-100. When these values are not sufficiently low, then credit is considered (along with impact to the SSC classification and DBA analyses) for retention in the fuel kernel, fuel pebble matrix, and reactor building and reevaluated in validated models as described in specific LBE analyses in Chapter 3.

A.3. Radionuclide Transport in the Primary Circuit and Reactor Building

Once released from the matrix, fission products will circulate with the primary helium coolant, subject to plateout and liftoff to and from surfaces throughout this system. The behavior of fission products is complicated by the presence of dust, generated by friction between pebbles in the core and as they pass through the fuel handling system. The dust particles may remain attached to pebble surfaces (and therefore move through the core and fuel handling system along with the pebbles) or be entrained into the flowing helium. The dust particles themselves are subject to deposition and resuspension processes, and fission products may plateout or liftoff from the dust particles wherever they reside. There are therefore five distinct radionuclide inventories (all spatially varying) that must be accounted for in the overall radionuclide plant balance:

- Radionuclides circulating as vapors in helium
- Radionuclides plated out directly on surfaces
- Radionuclides attached to dust circulating in helium
- Radionuclides attached to dust attached to pebbles
- Radionuclides attached dust deposited on surfaces

The transport phenomena governing this distribution are those that transport radionuclides from one of these inventory groups to another. These can be grouped into three categories of opposing processes:

- Plateout and liftoff of radionuclide vapors to/from (metal) surfaces.
- Deposition and resuspension of dust particles themselves, on/from pebbles or other primary surfaces.
- Adsorption and desorption of radionuclides on/from dust particles. This may occur wherever the dust resides, (i.e., on pebbles, other surfaces, or while circulating in the helium).

These are the essential phenomena governing radionuclide transport within the HPB, and the relationships between the inventories and transport phenomena are illustrated in Figure A-2. Here the inventories are represented by green boxes and the transport phenomena by yellow boxes; red arrows trace transport paths leading to greater release, and blue arrows those pathways leading to greater retention.

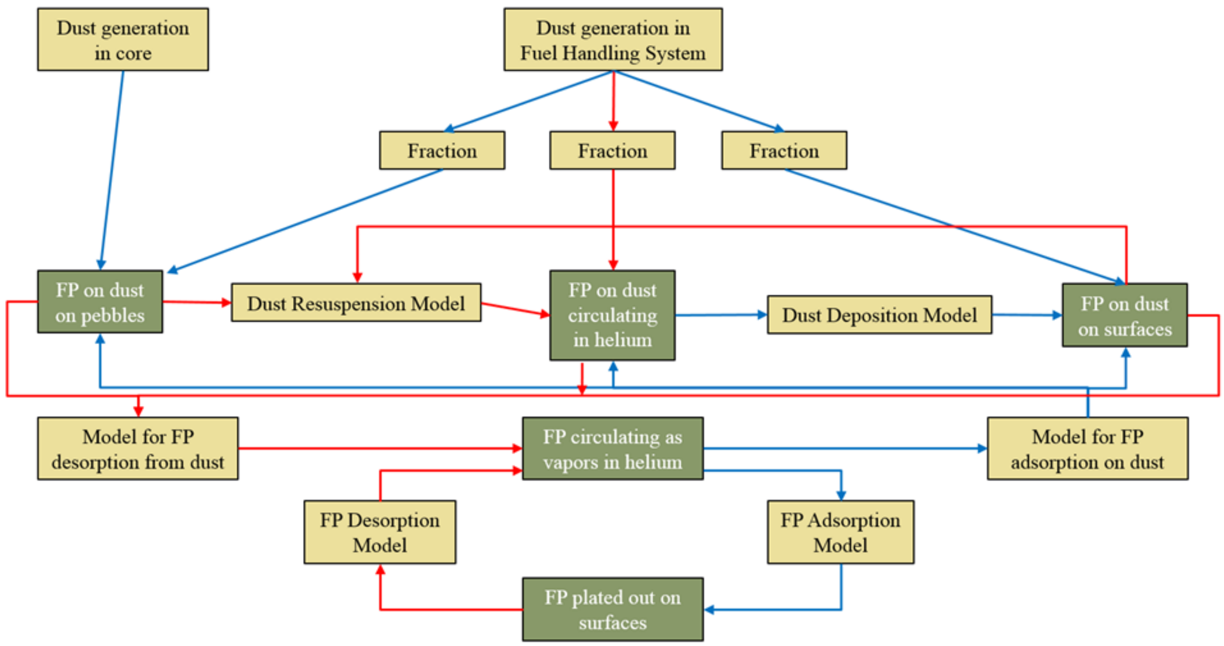

Figure A-2. Radionuclide Transport Phenomena in the Primary Circuit and Reactor Building

In the event of a helium pressure boundary breech and depressurization, radionuclides and radionuclide-laden dust that are circulating in the helium may be released to the reactor building, and the associated blowdown of the primary circuit and resultant flow transient may increase liftoff/resuspension/desorption processes occurring there. While the conditions are certainly different (different gas or gas mixture, lower temperatures, pressures, and flow rates), essentially the same set of transport phenomena are active in the reactor building (obviously excluding any interactions with pebbles), though the plateout/deposition/adsorption processes can be expected to predominate there. Any radionuclides not retained via such processes are potentially released to the environment.

A.4. XSTERM Code Suite Overview

X-energy developed the MST code development XSTERM as part of an approach to quantify the Xe-100 source terms. The phenomena that need to be modeled to predict MSTs for normal and postulated accidents are discussed in [Chapter 3, Section numbers] and illustrated in Figure A-2.

The code suite employed in the MST calculation chain consists of code modules for:

- Design and transient analysis (e.g., structures, reactor physics, thermo- and fluid-dynamics)
- Overall radionuclide plant mass balance consistent with the radionuclide design criteria
- TRISO-coated fuel particle performance
- Radionuclide production, decay, transmutation and transport in the fuel sphere
- Radionuclide retention, transport and release from fuel spheres into the helium coolant
- Transport and distribution of radionuclides, including dust effects, within the HPB

- Transport of radionuclides into the reactor building and to the environment for those accident scenarios that entail a depressurization of the helium pressure boundary

X-energy's MST code suite, XSTERM, includes a number of code modules that operate in a chain fashion to evaluate the source term and subsequent dose consequences of Xe-100 evaluation models of event sequences. Each code module has a specific modeling scope.

[A series of descriptive tables would be included to further present details on specific code module capabilities and comparable codes.]

A.5. Transient and Safety Analysis Methods

The Xe-100 was developed using several important codes and software packages to conduct design, transient, and safety analyses. The evaluation of Design Basis Accidents (DBAs), which only rely on safety-related SSCs to perform their required safety functions, make use of industry-standard codes and software implemented in a quality assured manner through X-energy's Quality Assurance Program Description [XEQAPD 1.0]. Evaluation models are developed for event sequences in accordance with the guidance of RG 1.203 as described in the Xe-100 topical report "Transient and Safety Analysis Methodologies" [XE00-R-R1ZZ-RDZZ-L-000714], which is a general reference. Details of specific evaluation models, their acceptance criteria, and analysis methods are described in that report. A summary of specific evaluation models, analysis methods, and codes/software used in the analysis is provided here. Code-specific verification and validation activities are captured in individual topical reports as described later. The general design approach to integrating simulation and modeling is shown in Figure A-3.

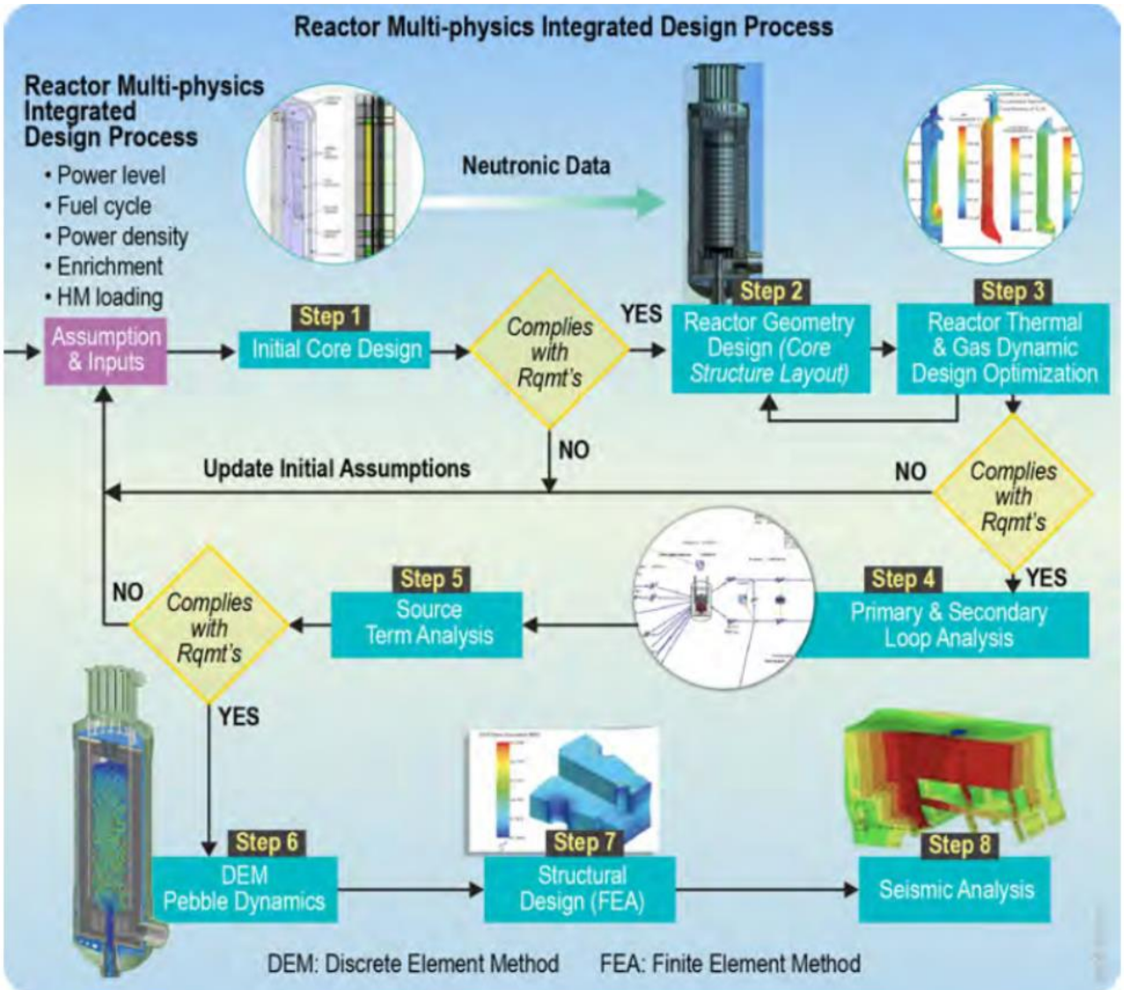

Figure A-3. Integrated Simulation Toolset Used in the Xe-100 Design

A.5.1 Structural Analysis of Civil Structures

[Not exercised yet]

A.5.2 Piping and Pressure Vessel Analyses

[Not exercised yet]

A.5.3 Thermal-Hydraulic Analyses

[Not exercised yet]

A.5.4 Neutronics Analyses

[Not exercised yet]

A.5.5 Criticality Analyses

[Not exercised yet]

A.5.6 Integrated Plant Response Analyses

[Not exercised yet]

A.5.7 Electrical Transient Analyses

[Not exercised yet]

Appendix B Draft Content for SAR Chapter 8 – Plant Programs

This chapter describes the purpose, scope, and performance objectives of plant programs that support the Xe-100 safety case. These programs and their elements are relied upon to provide reasonable assurance that the reliability and availability targets of safety-related (SR) and Non-Safety-Related with Special Treatment (NSRST) Structures, Systems, and Components (SSCs) are met through the plant lifecycle, and that safety-significant uncertainties are effectively addressed as part of the defense-in-depth (DID) adequacy evaluation described in Chapter [4].

The Xe-100's Plant Programs are organized by lifecycle phase. These phases are:

1. Design
2. Construction
3. Commissioning and Start-up
4. Operations

Where applicable, both quantitative and qualitative performance objectives of the program are described in order to provide sufficient bases for evaluation of the adequacy of each programmatic element to fulfill the special treatment requirements for associated SR and NSRST SSCs or to provide adequate DID. The detailed implementation of each program is contained in various programmatic documentation (i.e., processes, procedures, manuals, program records) maintained by the applicant and its contractors. When a program has previously received review by the NRC staff, the associated topical report and safety evaluation are cited and incorporated by reference.

B.1. Design Phase Plant Programs

B.1.1 Quality Assurance Program

During the Design phase, X-energy maintained quality assurance through a comprehensive, integrated management system as described in XEQAPD 1.0 "Quality Assurance Program Description," Rev 3-A [Ref xyz]. This topical report was reviewed by the NRC staff as documents in the included Safety Evaluation dated September 2020. Both the QAPD and SE are incorporated by reference in this application.

The X-energy QAPD is based on the guidance of Regulatory Guide 1.28, "Quality Assurance Program Criteria (Design and Construction)" Revision 5 [Ref xyz] and implements the quality assurance elements of ASME NQA-1-2015 "Quality Assurance Requirements for Nuclear Facility Applications" [Ref xyz]. Departures from the NQA-1-2015 standard are described in X-energy's QAPD.

The QAPD describes how the elements of 10 CFR 50, Appendix B, are implemented through the Design Phase of the Xe-100 in order to meet applicable regulatory requirements. These elements are evaluated for each SR SSC to determine applicability based on the specific requirements, including the grading of requirements when applicable.

For NSRST, a graded approach to quality assurance is applied to ensure the reliability and capability of each NSRST SSC is achieved. The application of the graded approach is described in Appendix D of X-energy's QAPD.

B.1.2 Reliability, Availability, and Capability Assurance Program

Each SR and NSRST SSC has associated reliability, availability, and capability targets. These targets were developed through an iterative process within the Licensing Modernization Project (LMP) framework. Initially, estimates from trade studies, expert elicitation, and operating experience were used in the probabilistic risk assessment to identify, categorize, and select Licensing Basis Events (LBEs) for further evaluation. Throughout this evaluation process, the targets associated with SSC reliability and availability to perform their associated required safety function were captured and established in the Reliability, Availability, and Capability Assurance Program (RACAP) for lifecycle management. Targets that are identified in the Design Phase are flowed into procurement and construction processes in the Construction Phase, confirmed (when possible) in the Commissioning and Start-up Phase, and maintained through the plant's lifetime through an Operations Phase Reliability and Integrity Management Program.

The primary purpose of the Design Phase RACAP is to ensure reliability, availability, and capability targets are identified and documented for all SR and NSRST SSCs, the basis for those targets is established, and the targets are transferred through to future phases for lifecycle management.

Where the RACAP is used as a special treatment to maintain the reliability and availability targets, it is identified in Chapter 6 and 7 tables for SR and NSRST SSCs.

The detailed implementation of X-energy's RACAP is described in XE00-R-R1ZZ-RDZZ-L-000xxx "Xe-100 Technical Report: Reliability, Availability, and Capability Assurance Program" Revision 0, dated January 3, 2021 [Ref xyz]. This controlling program document is maintained by X-energy and forms the basis for the Construction, Commissioning and Start-up, and Operations Phase programs.

B.1.3 Technical Specifications

The Xe-100 Technical Specifications are provided in the application in Part [ ]. Technical Specification safety limits were derived from the RSFs and RFDC described in Chapter [ ] of this SAR. The use of limiting conditions for operation, limiting safety system settings, establishment of design features to protect inherent safety function accomplishment, and surveillance requirements are all used to provide ongoing programmatic DiD for the plant.

B.2. Construction Phase Programs

[Not exercised yet]

B.3. Commissioning and Startup Phase Programs

[Not exercised yet]

B.4. Operations Phase Programs

[Not exercised yet]