Southern Company

Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors

High Temperature, Gas-Cooled Pebble Bed Reactor Licensing Modernization Project Demonstration

Project Report
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List of Abbreviations

ANS	American Nuclear Society
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
BDBE*	Beyond Design Basis Event
CFR	Code of Federal Regulations
CHG	Control Heat Generation
CR	Control Rod Withdrawal
CT	Circulator Trip
DBA	Design Basis Accident
DBE*	Design Basis Event
DID	defense-in-depth
DOE	Department of Energy
F-C	Frequency-Consequence
FW	Steam Generator Feedwater Pump Trip
HPB	Helium Pressure Boundary
HTGC-PBR	High Temperature, Gas-Cooled, Pebble Bed Reactor
HTGR	high temperature gas-cooled reactor
IAEA	International Atomic Energy Agency
IDP	Integrated Decision Panel
IE	Initiating Event
IPS	Investment Protection System
LBE*	Licensing Basis Event
LD	Large Helium Depressurization
LF	Loss of Primary Flow
LO	Loss of Offsite Power
LMP	Licensing Modernization Project
MD	Medium Helium Depressurization
MHTGR	a specific modular high temperature gas-cooled reactor designed by General Atomics
non-LWR	non-light water reactor
NPP	nuclear power plant
NSRST	Non-Safety-Related with Special Treatment
NSR	Non-Safety-Related with No Special Treatment
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
PBMR	Pebble Bed Modular Reactor
PRA	Probabilistic Risk Assessment
QHO	Quantitative Health Objective
RB	Reactor Building
RCCS	Reactor Cavity Cooling System
RCH	Remove Core Heat
RIPB	risk-informed and performance-based
RPS	Reactor Protection System
RT	Reactor Trip
SD	Small Helium Depressurization
SG	Steam Generator Tube Rupture
SR	Safety-Related
SSC	Structures, Systems, and Components
SU/SD	Start-up/Shutdown
TT	Turbine Trip
U.S.	United States

*These terms have special meanings defined in this document.

1.0 INTRODUCTION

The High Temperature, Gas-Cooled, Pebble Bed Reactor (HTGC-PBR) Demonstration Project described in this document directly supports the Licensing Modernization Project (LMP), a Southern Company-led, Department of Energy- (DOE-) supported effort to achieve the desired technology-inclusive and risk-informed and performance-based (RIPB) pathway to licensing of advanced non-light water reactors (non-LWRs). The Project's purpose, scope, objectives, and deliverables represent products supporting the LMP effort. These were developed in the project's initial charter and are summarized below.

1.1 Purpose

The purpose of the HTGC-PBR Demonstration Project was to exercise key processes as described in the LMP Guidance Document.[1] Each of the constituent components of the Guidance Document process had been employed in previous DOE and industry initiatives with positive results; the Demonstration performed with X-energy was the first opportunity to implement the process in the form of the draft LMP Guidance Document. Given the previous DOE and industry work which is foundational to the LMP, it was <u>not</u> the purpose of the HTGC-PBR Demonstration Project to determine whether the proposed process is feasible to implement or to justify the process by producing particular results – affirmative answers to those questions have long been observed and documented. The output of the Demonstration was used to improve the regulatory certainty of X-energy's Xe-100 design and its associated safety design approach. Additionally, output of the Demonstration provided insights to the Xe-100 design-specific regulatory strategy.

1.2 Scope

The design basis used for this HTGC-PBR Demonstration was defined as X-energy's high temperature pebble bed design augmented with similar designs and incorporating a Phase 0 probabilistic risk assessment (PRA). The Xe-100 design project has completed the pre-conceptual design phase and is currently midway through the conceptual design phase. The basis for the Xe-100 design for this project is described in a "Pre-Conceptual Design Basis Report."[2] Section 3.3 of this document includes a summary of major design selections that resulted from trade studies and analyses undertaken during pre-conceptual design. The demonstration utilizes relevant information from gas-cooled reactor designs that completed conceptual design and underwent preapplication reviews by the NRC: the modular high temperature gas-cooled reactor designed by General Atomics (MHTGR),[3] the Pebble Bed Modular Reactor (PBMR),[4] and the Next Generation Nuclear Plant (NGNP).[5] Additionally, the Demonstration incorporates a Phase 0 PRA completed by X-energy.[6] Given the current state of the design and available information, the Phase 0 PRA was limited in scope and level of detail to that of the pre-conceptual design and includes only internal Initiating Events (IEs) with the four-module plant operating at full-power. The PRA systematically documented Xe-100 design assumptions for each of 11 internal event sequences.

The LMP process guidance used in this HTGC-PBR demonstration was the Guidance Document. With this being a demonstration, specific portions of the Guidance Document were demonstrated such as selection and quantification of Anticipated Operational Occurrences (AOOs), Design

Basis Events (DBEs), Beyond Design Basis Events (BDBEs) (reference Section 3.0 of the Guidance Document) and required safety functions, and candidate Structures, Systems, and Components (SSCs) safety classifications (reference Section 4.0 of the Guidance Document).

1.3 Objectives

The objectives of the HTGC-PBR Demonstration were:

- Demonstrate key processes within the LMP Guidance Document to X-energy.
- Leverage the LMP process to improve the regulatory certainty of X-energy's Xe-100 design and safety case, as best possible at the current state of design, by identifying a credible spectrum of Licensing Basis Events and investigating available structure, system, and component (SSC) groupings that result in acceptable outcomes for the identified LBEs. Underlying this objective is the assertion that use of risk informed, performance based methods to reach these conclusions are endorsed by Commission policy and compatible with the existing regulatory framework.

1.4 Deliverables

The deliverables of the Demonstration described in this document include:

- Licensing Basis Event (LBE) selection and quantification, involving AOOs, DBEs, and BDBEs
- Identification of the Required Safety Functions
- Safety Classification of SSCs

The Demonstration also included a brief introduction to X-energy of the LMP approach for RIPB evaluation of defense-in-depth (DID) adequacy. Given the current state of the X-energy reactor design, the DID portions of the draft Guidance Document were not exercised at this time.

2.0 BACKGROUND AND LINKAGE TO LMP

The U.S. commercial nuclear power industry has long sought a broadly applicable, NRC-accepted, RIPB licensing framework. Incremental advances, accelerated recently by increased interest in licensing advanced non-LWRs and Congressional interest, have resulted in an opportune time to pursue NRC endorsement of a RIPB framework. That framework is being advanced currently by the LMP, a Southern-led, DOE-supported effort to achieve the desired technology-inclusive, RIPB pathway to licensing of non-LWRs specifically. This Demonstration is an applied execution of the RIPB processes proposed by the LMP.

2.1 LMP Guidance Document

The LMP is currently developing a stand-alone Guidance Document describing the LMP processes for PRA development, selection, and evaluation of LBEs, SSC safety classification and performance requirements, and evaluation of DID adequacy. The Guidance Document extracts important regulatory insights from a series of documents covering the same topics which describe the technical bases for the performance of RIPB decisions associated with designing and licensing advanced non-LWRs. The Guidance Document is intended to be endorsed by the NRC in the form of a Regulatory Guide for licensing advanced non-LWRs and is planned for release in early 2019.

2.2 LMP Documents

Probabilistic Risk Assessment Approach
The PRA draft white paper approach document contains the historical background, technical justifications and supporting information, and implementation guidance for creating a PRA computer model fit for providing insights into plant behavior for a given phase of design development. The PRA approach is reactor technology inclusive and makes use of technology inclusive risk metrics. The PRA is introduced at an early stage of design to incorporate risk insights into early design decisions. The PRA models are initially limited in scope and of a coarse level of detail as constrained by available supporting information. The scope and level of detail of the PRA models are increased as design and site information are available. The RIPB decisions supported by the PRA and deterministic safety approaches are reviewed and revised as the risk model definition is brought into focus. This document is available as Reference [7].

Selection of Licensing Basis Events
Key to building the safety case of any reactor design is identifying, selecting, and evaluating LBE, including the Design Basis Accidents (DBAs). The LMP proposed approach is designed to identify LBEs that reflect the reactor design and technology specific issues and challenges associated with each reactor's safety design approach. A systematic and prescriptive process is used to determine the safety functions required to meet risk targets, whose process provides the developer with options to select the safety-related SSCs that will be used to demonstrate satisfaction of requirements for the Design Basis Accidents. This process builds on the PRA model and is tightly linked with the safety classification of SSCs. This white paper document is available as Reference [8].

Safety Classification and Performance Criteria for Structures, Systems, and Components

Criteria are provided to classify SSCs into three safety classes: SSCs are either Safety-Related (SR), Non-Safety-Related with Special Treatment (NSRST) or Non-Safety-Related with no Special Treatment (NSR). Based on the SSC safety functions in the performance of both prevention and mitigation functions, the developer assigns reliability and performance targets which help ensure that selected special treatment requirements are performance-based. This LMP white paper document is available as Reference [9].

Risk-Informed and Performance-Based Evaluation of Defense-in-Depth Adequacy

The concept of DID has long been an expressed philosophy of commercial nuclear power design, licensing, and operation. This LMP white paper proposal document seeks to systematically evaluate DID adequacy for the plant capabilities and programs that comprise DID, incorporate needed layers of defense to address uncertainties in the design and operation of the plant, and establish a fixed baseline of DID adequacy. This document is available as Reference [10].

3.0 DEMONSTRATION OVERVIEW

3.1 Summary of Demonstration Activities

The Demonstration progressed through the traditional project management phases: initiation, planning, execution, monitoring and controlling, and closeout. During the planning phase, a cross-functional, multi-company core team consisting of Southern Company, X-energy, and various industry experts was assembled. This team included subject matter experts on PRA, RIPB processes, technical project execution, and LWR fleet operations. A project charter including objectives and deliverables was developed and subsequently approved by the LMP. The core team began executing the project charter by completing significant preparatory work which culminated in a week-long face-to-face workshop at X-energy facilities in Greenbelt, MD, in April 2018. X-energy personnel and their direct contractors performed the pebble bed HTGR PRA analysis, consequence analyses, and provided design-specific engineering insights; Southern Company and industry consultants provided project organization, guided the execution of the LMP RIPB process demonstration of the pebble bed HTGR, and led authoring of this report. The workshop included daily working sessions which executed steps in the LMP processes for the Xe-100 pre-conceptual design and included daily interaction with X-energy staff responsible for producing and evaluating the conceptual design of the Xe-100. The outputs, lessons, and conclusions from this effort are part of the closeout phase and are included in this document.

3.2 Prerequisites and Inputs for the Demonstration Project

The PRA utilized for the HTGC-PBR Demonstration was developed during the pre-conceptual Xe-100 design phase, and is referred to as the Phase 0 PRA.[6] The Phase 0 PRA relied upon work and knowledge from previous high temperature gas-cooled reactor (HTGR) design projects and incorporated pre-conceptual Xe-100 design descriptions[2] to provide a best-estimate of the event sequences and their associated frequencies. The purpose of the Phase 0 PRA was to provide risk insights for the conceptual design development. As specified in the LMP PRA document, the goal of the PRA in the LMP approach is to support risk-informed decisions associated with each stage of design and licensing. The scope and level of detail of the PRA will be developed in stages and correspond to the scope and level of detail in each stage of design and licensing.

Given the current state of the design and available information, the Phase 0 PRA was limited in scope and level of detail to that of the pre-conceptual design and includes only internal IEs with the four-module plant operating at full-power. As a result of this, the tasks to identify the design basis external events and to finalize the selection and evaluation of LBEs are deferred until greater design completion is achieved. However, even in this early phase of the design, the PRA included event sequences involving a single reactor module as well as sequences involving multiple reactor modules in the four-reactor module plant. This section provides a description of some of the major inputs to the Phase 0 PRA development and key aspects of the PRA that are critical to the HTGC-PBR Demonstration.

3.2.1 Early PRA Development for HTGRs

HTGR PRA has a long development history, and the projects which most strongly influenced the Phase 0 PRA included: MHTGR,[3] PBMR,[4] NGNP,[5] and first-hand knowledge from previously constructed and operated German pebble-bed reactors. The NGNP PRA played a significant role in helping define the Phase 0 PRA elements to be consistent with the applicable standards and regulatory guides. In-lieu-of plant design hazard analyses, the systematic search for IEs, safety function development, and event sequence modeling framework for the construction of the Phase 0 PRA was directly related to the NGNP and MHTGR PRAs. Both the MHTGR and PBMR PRAs played a critical role in defining similar pre-conceptual Xe-100 system models, their failure modes, and IE frequencies.

3.2.2 Pre-Conceptual Xe-100 Design Documents

The major pre-conceptual Xe-100 design documents used to facilitate the development of the Phase 0 PRA are described in the Pre-Conceptual Design Basis Report,[2] which contains a summary of the available trade studies and analysis reports that provide the basis for the pre-conceptual design selections. These design selections were then utilized to construct a specific, pre-conceptual Xe-100 PRA, built on fundamentals from prior HTGR designs. Although the PRA development benefitted from previous HTGR PRAs, the Phase 0 PRA incorporates important design features not in previous HTGR designs. These features include a unique concept for alternative forced reactor cooling that avoids the need for a separate shutdown heat removal system and a novel approach to detect and mitigate steam generator tube leaks.

3.2.3 Phase 0 PRA for Pre-Conceptual Xe-100 Design

The Phase 0 PRA, documented in an X-energy report,[6] is currently being upgraded to be consistent with the Xe-100, as it progresses through the conceptual design phase. The stated objectives of the Phase 0 PRA are to:

- Identify the Xe-100 nuclear power plant (the full plant including the nuclear island and the balance of plant) SSCs that need to be considered in the PRA development
- Identify open questions and potential issues about the existing reference design features and the capabilities of the SSCs in the prevention and mitigation of event sequences
- Define a set of IEs and event sequences that help to frame risk insights about the capabilities to prevent and mitigate accidents and to select the safety-related SSCs
- Identify possible event mitigation strategies to be considered in the Xe-100 design development and key assumptions that need to be resolved in the next phase of the design
- Provide preliminary estimates of the frequencies of selected IEs and event sequences and a qualitative characterization of the event sequence end states and their radiological consequences, yielding a preliminary list of LBEs to begin the Xe-100 conceptual design

Mechanistic source term (MST) and radiological consequence analysis (RCA) are integral elements of the ASME/ANS Non-LWR PRA standard. However, the evaluation of event sequence consequences in the X-energy pebble bed HTGR Phase 0 PRA was based on a

qualitative end state code assignment that determines whether the sequence involves a release of radionuclides, and, if so, the key factors along the sequence that determine the radionuclide transport pathway and magnitude of the release. The reliance on qualitative end state codes is primarily due the large number of event sequences, each with only slight variations in expected offsite dose calculations. Additionally, the MST and RCA tools needed to calculate the magnitude and timing of the releases specific to the Xe-100 reactor and fuel design, are not currently available for the Phase 0 PRA. For later Xe-100 design and PRA updated phases, it is expected that these tools will be available.

Fortunately, a large body of prior source term analysis for HTGRs exists in the public domain. This body of work was utilized for allocating consequences to the Phase 0 PRA event sequences and end states as starting point for the HTGC-PBR demonstration project. LBE offsite dose consequence values for plotting LBEs against a Frequency-Consequence Target is a prerequisite for all the proposed risk-informed LMP processes.

The Phase 0 PRA development follows similar process steps as those described in the LMP PRA document.

3.3 LBE Selection

The LBE selection process follows the approach described in the LMP LBE selection document and Guidance Document. The proposed Frequency-Consequence (F-C) target used for the HTGC-PBR Demonstration is shown in Figure 1. A design objective of the Xe-100 is to keep the LBEs well within the F-C target such that the resulting margins can support the eventual demonstration of DID adequacy.

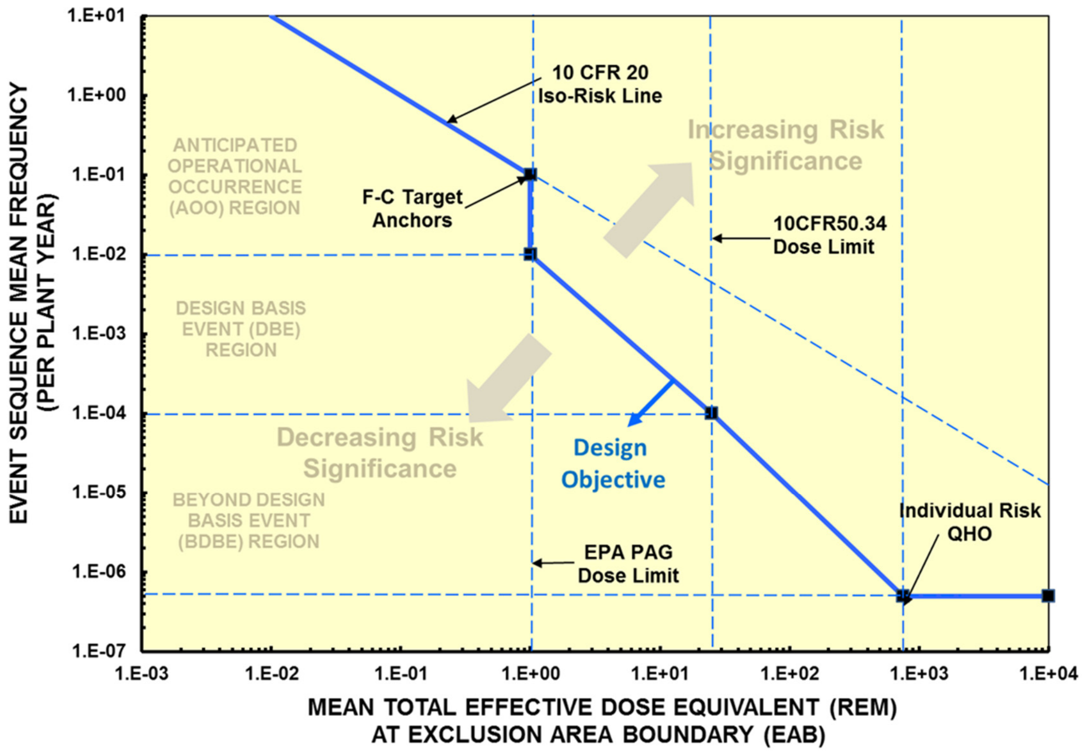

Figure 1. Frequency-Consequence Evaluation Criteria Proposed for LMP from Reference [1]

The LMP LBE selection and evaluation process is implemented as indicated in the steps on the flow chart shown in Figure 2.* The LBE selection and evaluation steps of Figure 2 that were addressed in the Demonstration include Steps 3, 4, 5a, 5b, 6, and 7a. Steps 1 and 2 had been performed for the pre-conceptual design prior to developing the Phase 0 PRA. In addition, the LMP approach for implementing Steps 7b, 7c, 7d, and 7e were reviewed with the X-energy design team, but were not implemented in the Demonstration for the Xe-100 design. Steps 8, 9, and 10 are beyond the scope of this HTGC-PBR Demonstration.

*One of the outcomes of the Demonstration was a recommendation to the LMP to revise this flow chart to provide more visibility to the task of identifying the required safety functions. This revised flow chart shown in Figure 2 has been incorporated into the latest draft of the LMP Guidance Document.

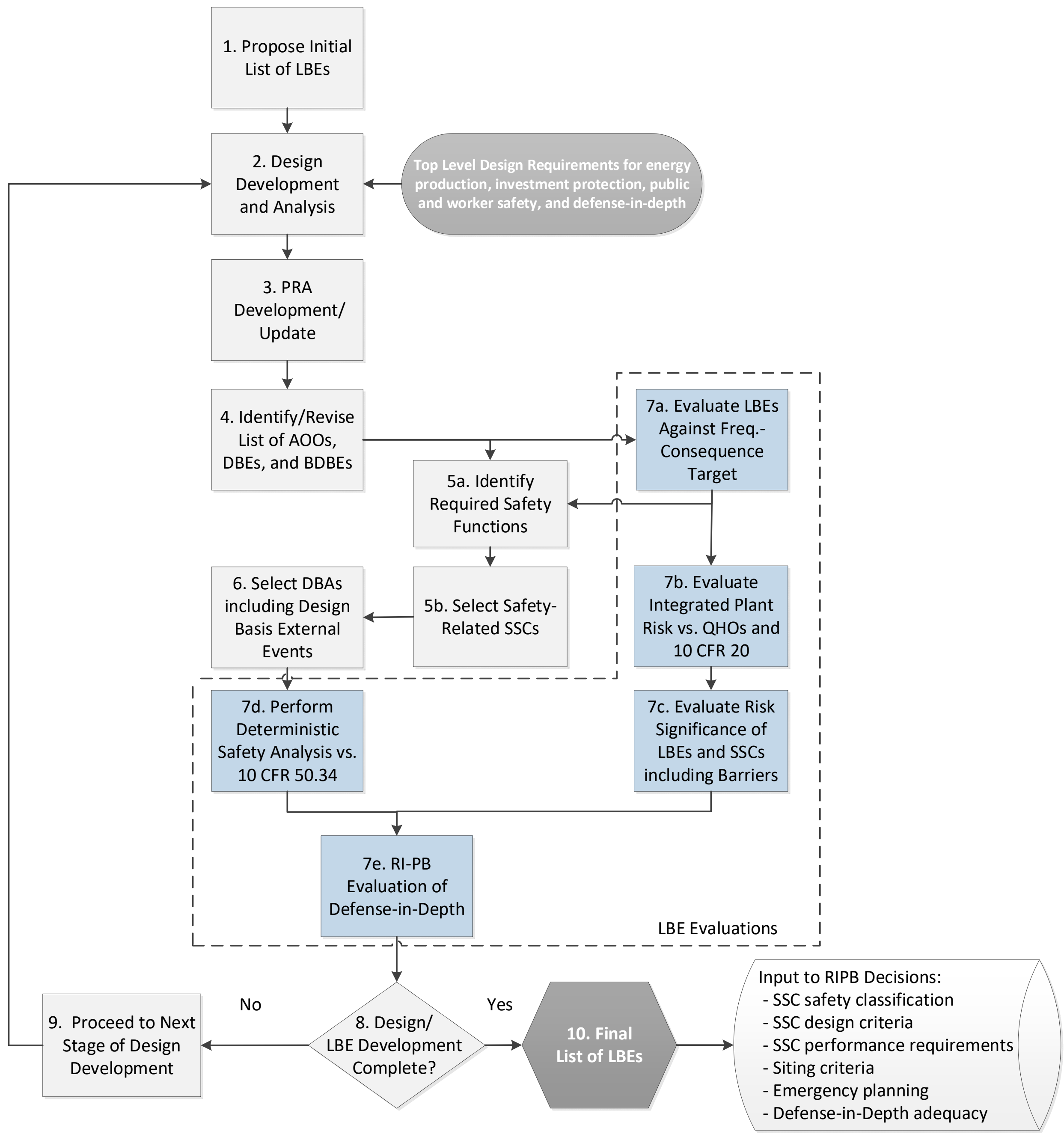

Figure 2. Process for Selecting and Evaluating Licensing Basis Events from Reference [1]

The LBE selection process begins with "Step 4. Identify/Revise List of AOOs, DBEs, and BDBEs." Then, "Step 7a. Evaluate LBEs Against Freq. vs. Consequence Target" is performed using the F-C target shown in Figure 1. Information from this step is then used in Step 5a to identify the required safety functions and in Steps 5b to select the safety-related SSCs that are available to mitigate the spectrum of DBEs for each required safety function. This information can then be used to define the DBAs in Step 6 by deterministically assuming that the DBEs rely only on the safety-related SSCs.

However, before the LBEs in Step 4 can be defined, it was necessary to develop consequence estimates for each of the LBEs, as this information is not available from the Phase 0 PRA. Per the Advanced Non-LWR PRA Standard,[11] this consequence data is to be developed using a mechanistic source term analysis and accompanying radiological consequence, dose, and dispersion analysis. It is the goal of the Xe-100 Phase 1 PRA to achieve this during the conceptual design phase. However, for the purposes of the HTGC-PBR Demonstration, consequence estimates are provided using information from previous HTGR PRAs.

The tasks performed in the Demonstration to support Steps 4 and 7a of LBE selection process in Figure 2 include the following:

- Identify and categorize the risk-informed LBEs from the Phase 0 PRA by event sequence frequency
- Estimate the consequences for the LBEs
- Evaluate the LBEs against the F-C target

3.3.1 Brief Background on the Phase 0 PRA and Identified LBEs

In order to discuss the identification of LBEs for the Xe-100, it is first necessary to review the background and key results of the Phase 0 PRA.[3] An important element of the PRA is the systematic search for 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.[11] However, for the current scope and available design information, IEs for the Phase 0 PRA are appropriately limited to internal events at full power. Consistent with the LMP PRA approach, consideration of external events and other modes other than full power is deferred to after the internal events at a future point in time when the design has progressed to such a detail that those analyses can be performed and produce actionable results for the designer. To ensure that an exhaustive enumeration of IEs appropriate for the Xe-100 design is accomplished, a structured process known as the Master Logic Diagram method is used. Table 1 lists the internal IEs identified through the Master Logic Diagram.

Table 1. 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 3, 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. The frequency basis is events per multi-module plant year, which is called for in evaluation of LBEs against the LMP F-C target.

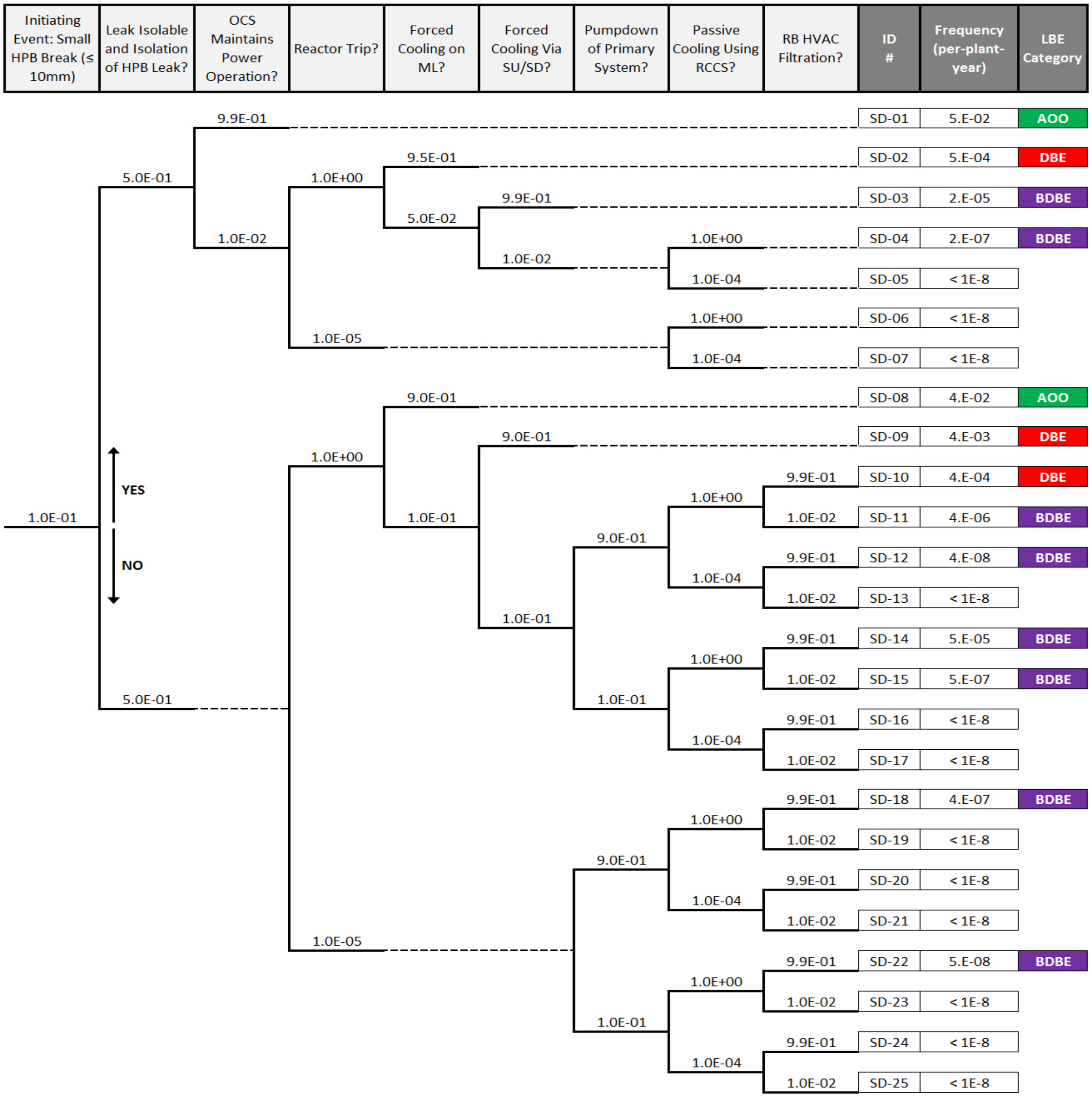

Figure 3. Small Helium Depressurization Event Tree with Associated LBEs

For the IEs in Table 1, a total of 11 AOOs, 17 DBEs, and 34 BDBEs are 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 depressurization and steam generator feedwater trip are those which involve a Helium Pressure Boundary (HPB) challenge. If the HPB is challenged, a radionuclide release into the Reactor Building (RB), and ultimately the environment is expected to occur. All other event sequences with an intact HPB do not lead to a release of radionuclides. For the Xe-100, components of a release of radionuclides are categorized, in order of ascending magnitude, as:

- Early release of Circulating Activity in the primary helium
- Early release of Plate out Activity on the primary internals
- Delayed Release from Fuel Element Activity
- Increased Release from above for water ingress events into the reactor

All sequences initiated by HPB leaks and breaks involve at least some or all circulating activity. For large breaks that involve large shear forces during depressurization, some lift-off of plate out may occur. If the sequence involves a loss of forced circulation, delayed fuel release may occur due to redistribution of core temperatures during the passive heat removal. For sequences involving a steam generator tube rupture with steam/water ingress to the primary system, a delayed fuel release may be enhanced due to graphite corrosion and fuel hydrolysis. In addition, some of the plate out may be washed off surfaces and be available for release. Depending on the sequence, there may be delayed fuel releases from the fuel but no release from the HPB if the relief valve on the HPB does not lift from the water/steam ingress and any reaction products.

3.3.2 Allocation of Consequences to the LBEs Identified by the Phase 0 PRA

Due to the lack of available mechanistic source term consequence data for the LBEs identified in the Phase 0 PRA, surrogate data is allocated to LBEs on an individual LBE basis, and scaled appropriately. Surrogate data applicable to most of the LBEs comes from the MHTGR and the PBMR PRAs.

While there are many factors to consider when scaling consequences from one reactor design to another, for a given LBE, scaling by core thermal power level is the most well understood and least subjective factor. Therefore, consequence values from the MHTGR are scaled by a factor of 0.57 (200 MWt/350 MWt), and from the PBMR by a factor of 0.75 (200 MWt/268 MWt). By not accounting for differences in other design factors such as initial primary pressure during a depressurization event and the physical break size differences in the deterministic calculations, among other factors, this leads to a more conservative estimate of the consequences. Accounting for additional factors would lead to a further reduction in the expected radionuclide releases.

However, given the increased subjective nature of accounting for further differences, the consequences are left as-is after scaling for core power. Additional modification of the consequences would likely be overridden by the large uncertainties associated with both the MHTGR and PBMR consequence evaluations. Furthermore, the principal focus of this Demonstration is not to compute accurate consequences for the LBEs, but rather to exercise the technology-inclusive, RIPB process on an HTGC-PBR.

Table 2 lists the LBEs that were identified in the Phase 0 PRA, a brief description of the event sequences, their estimated frequency, and allocated 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.

Table 2. LBEs for the HTGC-PBR Demonstration Grouped by Event Sequence Frequency

LBE	PRA ID	LBE Description	Event Sequence Frequency, per plant-yr	Dose, WB rem	Dose Basis, M/P
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	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), 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 < 3hr, 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	Md. Helium Break, isolation, forced cooling via SU/SD	5×10^{-4}	3×10^{-5}	P
10	SD-02	Sm. Helium Leak, isolation, forced cooling via main-loop	5×10^{-4}	1×10^{-5}	P
11	SD-10	Sm. Helium Leak, passive cooling via RCCS, pump down successful	5×10^{-4}	2×10^{-4}	M
12	MD-02	Md. 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), 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
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	Sm. 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	Md. 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}	$< 10^{-5}$	--

LBE	PRA ID	LBE Description	Event Sequence Frequency, per plant-yr	Dose, WB rem	Dose Basis, M/P
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	Sm. 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 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	Lg. Helium Break, passive cooling via RCCS, RB success	7×10^{-6}	8×10^{-5}	M
20	MD-02	Md. Break, isolation, forced cooling via SU/SD	6×10^{-6}	3×10^{-5}	M
21	SD-11	Sm. Helium Leak, no isolation, passive cooling via RCCS, pumpdown	6×10^{-6}	2×10^{-4}	P
22	MD-12	Md. 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	Md. Helium Break, no isolation, forced cooling via SU/SD	5×10^{-6}	2×10^{-4}	M
25	MD-03	Md. 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	Lg. Helium Break, passive cooling via RCCS, RB failure	2×10^{-6}	8×10^{-5}	M
28	LD-09	Lg. 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), 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 success	8×10^{-7}	8×10^{-4}	M
33	SD-15	Sm. Helium Leak, no isolation, passive cooling via RCCS	6×10^{-7}	4×10^{-4}	M
34	MD-15	Md. Helium Break, no isolation, passive cooling via RCCS	6×10^{-7}	2×10^{-4}	P

3.3.3 HTGC-PBR LBEs Plotted Against the LMP Frequency-Consequence Target

Using the LBEs and the allocated consequence values from Table 2, Figure 4 provides a comparison with the proposed F-C target.

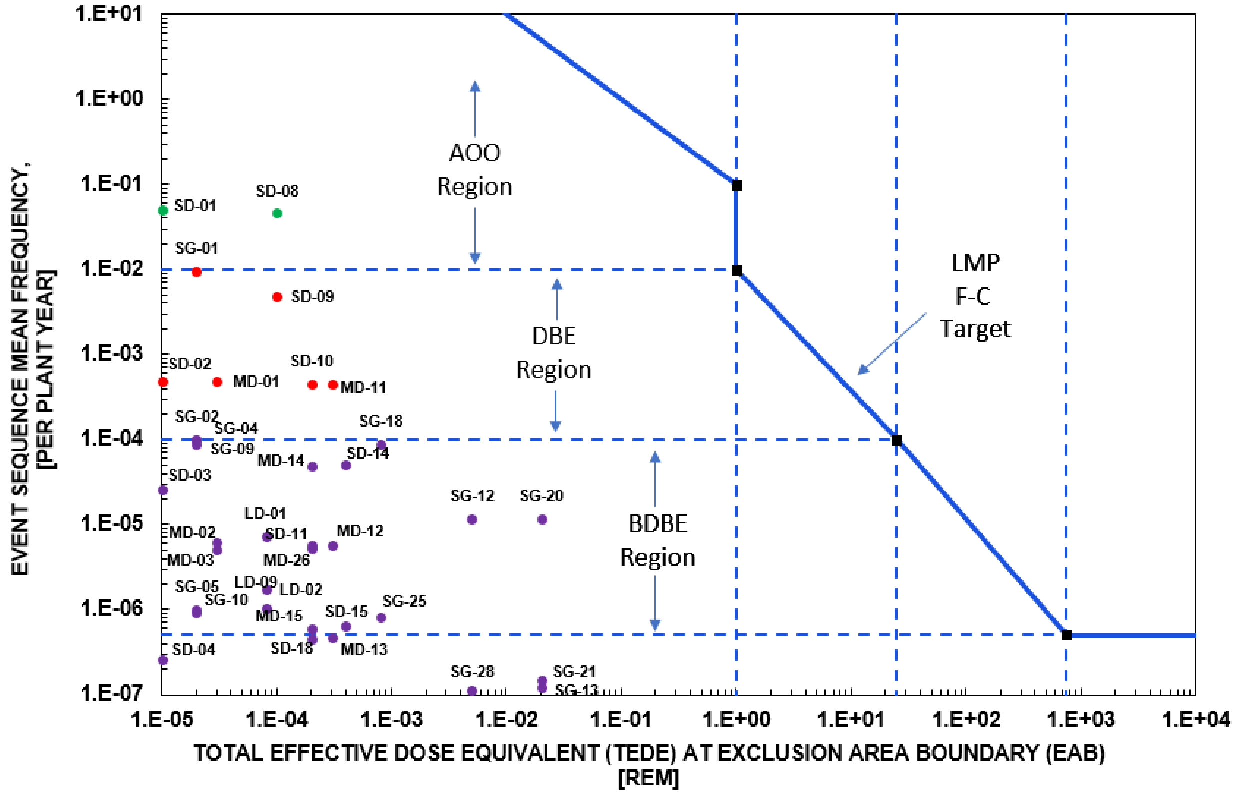

Figure 4. Xe-100 LBEs Plotted Against the LMP F-C Target

A key aspect to note from Figure 4 is that there are large margins between the LBE mean-value points and the target line. The mean-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, 1-rem Protective Action Guides criterion for sheltering and/or evacuation actions. Although outside of this demonstration, these margins are considered in the LMP evaluation of defense-in-depth adequacy step and will be covered in later phases of the design.

Another observation to note is that no uncertainty upper and lower bars are presented for any of the LBEs. No detailed uncertainty analysis was performed for the Phase 0 PRA in terms of frequency, and no uncertainty analysis in terms of consequence was performed for this HTGC-PBR Demonstration. However, given the large margin between the points and the target line, these LBEs including uncertainties are not expected to challenge the target line.

It should be noted that the current scope does not consider seismic events or other external hazards. Additional LBEs will be added to the F-C target evaluation as the design progresses. Movement of the currently identified LBEs will occur along both axes, as the design and analyses mature. Then each RIPB decision supported by the PRA will be reviewed and revised as needed to reflect new risk insights.

3.3.4 LMP Process Feedback for LBE Selection

The LMP LBE selection process as described in steps 4, 7a, and 7b of the Guidance Document was shown to be applicable to the Xe-100. 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.

3.4 Development of Required Safety Functions

This section derives the safety functions from the prior selection of LBEs as a necessary step in identifying the SSCs. Safety functions of SSCs are responsible for preventing and mitigating the release of radioactive material from any radionuclide source and help define the scope of SSCs to be modeled in the PRA. The general safety functions are hierarchical with the highest level applicable to nuclear power plants (NPPs) in general and the bottom level applicable to the HTGC-PBR of interest and to the Xe-100 LMP Demonstration. Figure 5 depicts these two levels.

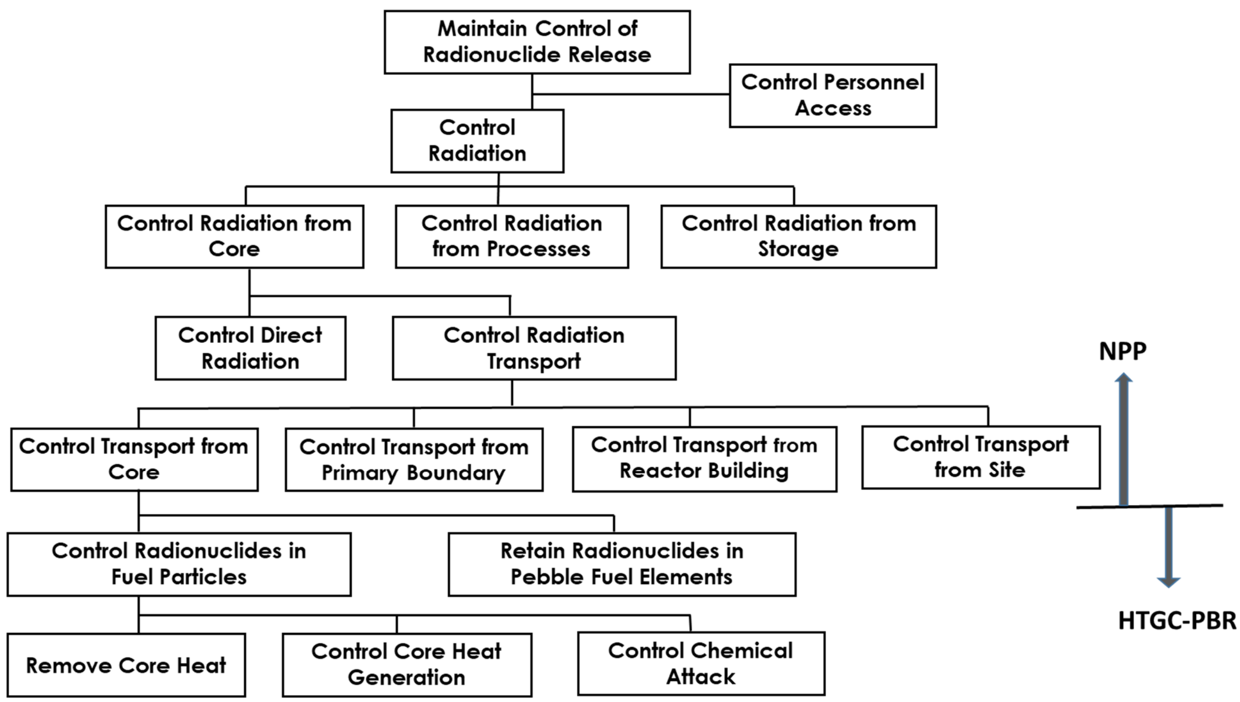

Figure 5. Nuclear Power Plant and HTGC-PBR Specific Safety Functions

Figure 5 includes protection of the personnel within the plant and the public off site. 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 offsite public from the reactor core are shown in the middle of the chart. At the HTGC–PBR lower level, the safety functions are expanded within the core, where the radionuclide retention is by design focused, specifically, within the ceramic coated fuel particles with the three key safety design functions shown at the bottom of the figure: Remove Core Heat, Control Core Heat Generation, and Control Chemical Attack.

The PRA and LBE selection documents referenced by the LMP Guidance Document each contain steps to identify the safety functions that protect the radionuclide barriers. The initial Xe-100 LBEs discussed in the prior section provide the basis to further examine the specifics of which functions must be met with margin to the F-C target to protect the public. Each event

sequence is examined to see which functions led to the acceptable radionuclide retention within the F-C target and, conversely, which sequences would not have met the consequence target if it were not for one or more SSCs in the design that performed a necessary safety function. For example, Figure 6 superimposes a hypothetical, unacceptable challenge. For all 17 of the DBEs from the 11 event trees discussed in Section 3.3.1, there are safety functions successfully performed by SSCs with capability to mitigate them to keep them to the left of the F-C curve and/or with reliability to prevent them so that they are in an acceptable BDBE region.

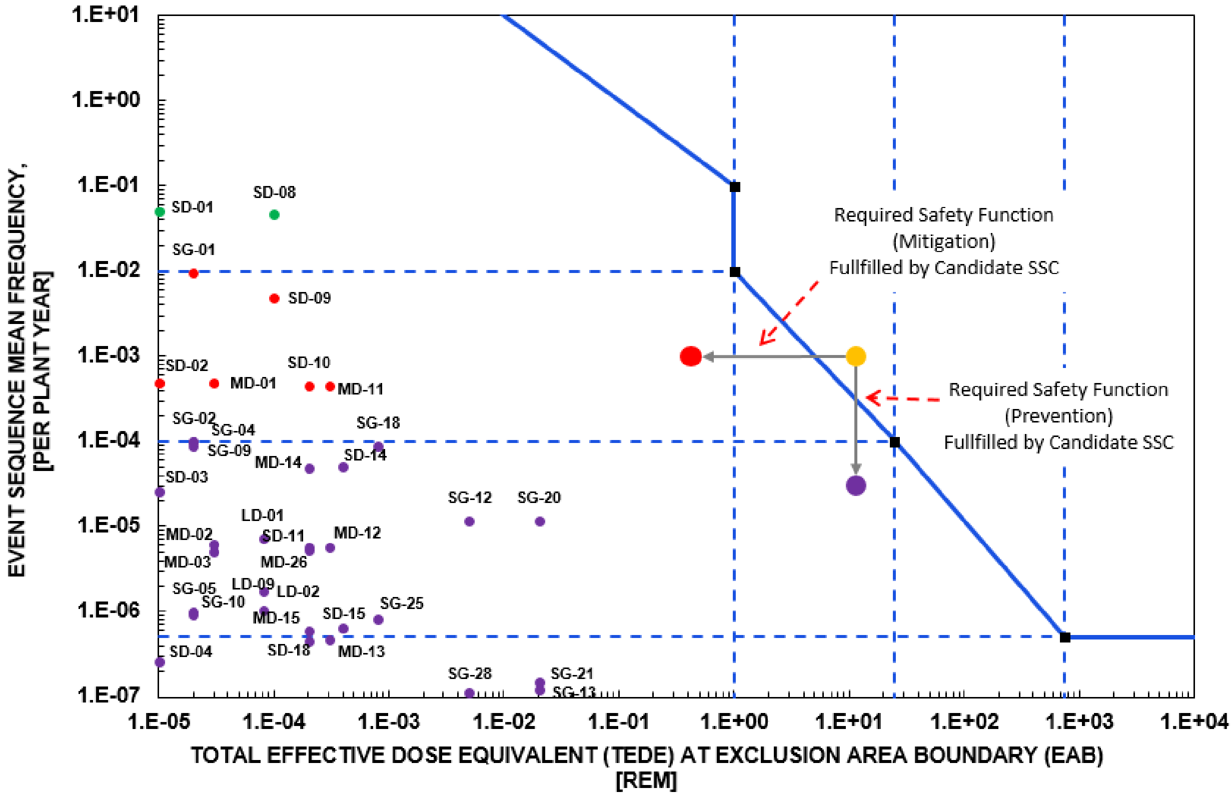

Figure 6. Concept of Safety Functions for Mitigation of DBEs and Prevention of High Consequence BDBEs

From this investigation, the subset of all the safety functions that are necessary and sufficient to mitigate DBEs and to prevent high consequence BDBEs to within the F-C target are determined. These required safety functions are shown shaded in blue in Figure 7.[9]

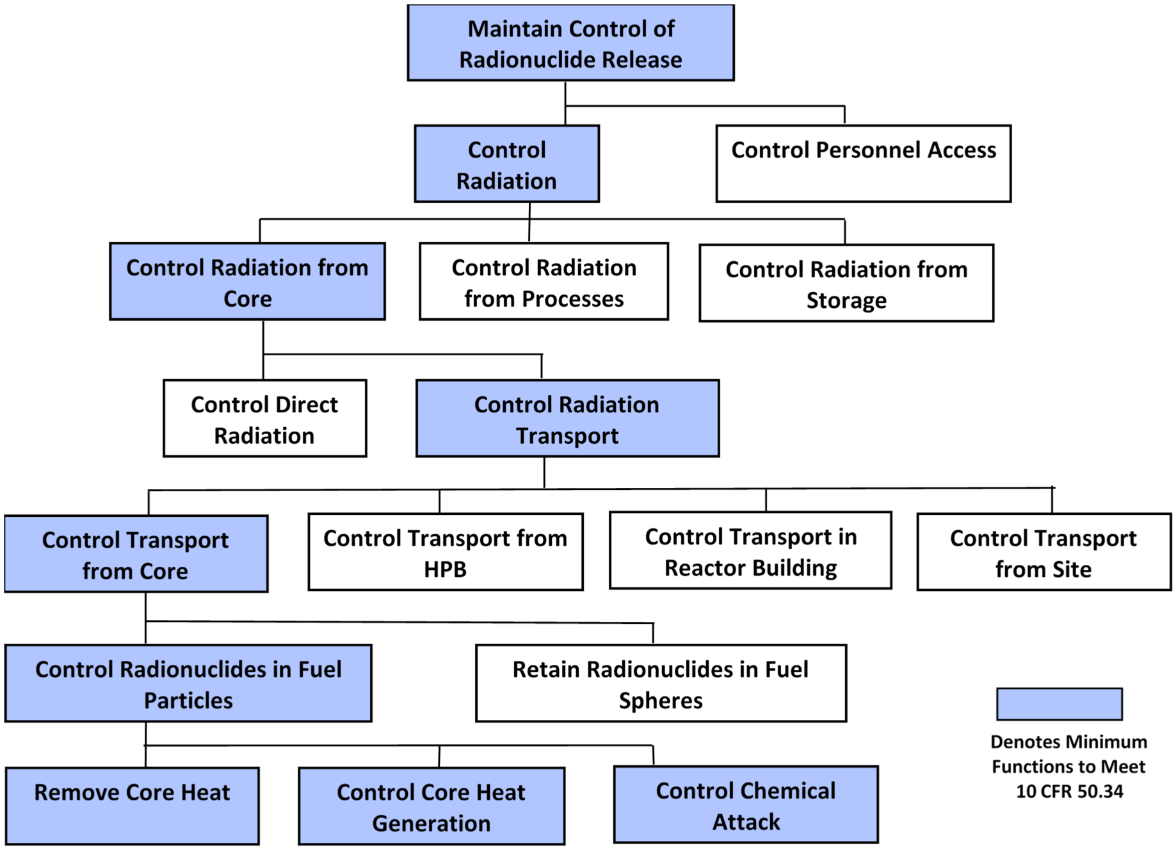

Figure 7. Required Safety Functions for the HTGC-PBR

3.5 Identification of SSC Safety Classification Options and Associated DBAs

The LMP process for SSC safety classification is described in the safety classification and performance criteria for SSCs and in the Guidance Document. The proposed process is systematic and reproducible and sufficiently complete to be applied to the current Xe-100 design, and is included in the HTGC-PBR Demonstration. The process identifies three categories of classification:

- Safety-Related
    - SSCs selected by the designer to perform required safety functions to mitigate the consequences of DBEs to within the F-C target, and to mitigate DBAs to meet the dose limits of 10 Code of Federal Regulations (CFR) 50.34 using conservative assumptions
    - SSCs selected by the designer to perform required safety functions to prevent the frequency of BDBEs with consequences greater than 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the F-C target
- Non-Safety-Related with Special Treatment
    - Non-safety-related SSCs relied on to perform risk significant functions. Risk significant SSCs are those that perform functions that keep LBEs from exceeding the

F-C target, or make significant contributions to the cumulative risk metrics selected for evaluating the total risk from all analyzed LBEs.

- Non-safety-related SSCs relied on to perform functions requiring special treatment for DID adequacy

- Non-Safety-Related with No Special Treatment
    - All other SSCs

This process outlines three critical steps after the required functions are identified. The first step is to determine which SSCs are selected to meet the required safety function for each DBE. These SSCs are then classified as safety-related. The next two steps involve determining if the SSC should be classified as NSRST. Due to limitations in the scope of the Demonstration, the focus was put on the selection of SR SSCs as illustrated in Figure 8. To exercise the process for classifying NSRST SSCs would have required a great deal of effort to perform the calculations needed to establish the risk significance of SSCs and to determine which SSCs perform functions that are required for DID adequacy. These tasks are deferred to a later stage of the Xe-100 design.

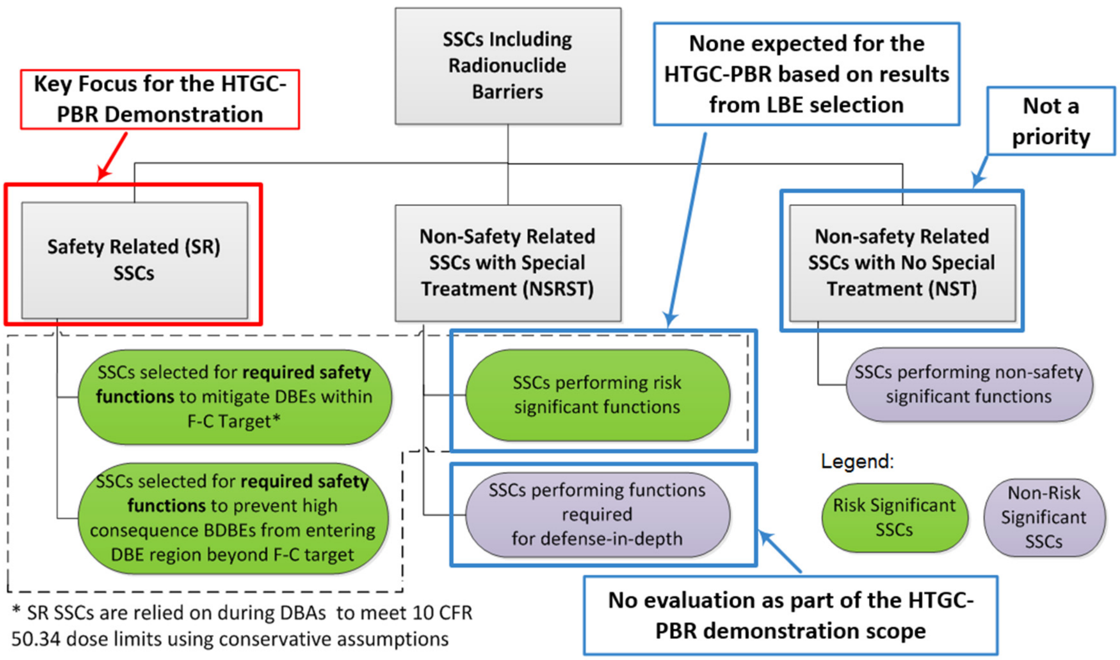

Figure 8. Annotated SSC Safety Categories for the HTGC-PBR Demonstration

Available sets of SSCs which could perform the three required safety functions are identified. The LMP process does not attempt to determine which of the options should be chosen, as this is a designer's choice. The designer may consider many different parameters when selecting the safety-related SSCs, such as economic cost, regulatory uncertainty, and difficulty of performance requirements. Therefore, the focus of the HTGC-PBR Demonstration is to identify the various options to meet the required safety functions, while leaving the specific choice of safety-related SSCs to the designer. However, to fulfill the LMP process, two example choices are made after all three safety functions are discussed.

3.5.1 Summary of the HTGC-PBR Demonstration Process for SSC Safety Classification

The process for identifying the safety-related SSCs for meeting each safety function is as follows:

1. LBEs with mean frequencies in the DBE region are evaluated for each required safety function.
2. Options (rows) are constructed, which contain a complete set of SSCs available to perform the required safety function.
3. These options are then evaluated as either being "YES" (Y) or available, or "NO" (N) or unavailable for each DBE.
4. If an option is unavailable for one or more DBEs, then this option alone is not carried forward as a candidate for the safety-related classification.
5. If an option is available for every DBE, then this option is presented as a candidate for being classified as safety-related.
6. This is performed for each safety function, and then each option is combined to construct a matrix of potential SSC options for being classified as safety-related.

At Step 6 the Demonstration stops for the purposes of identifying the SSCs available for being classified as safety-related. It is then up to the designer to choose from this matrix of options, after much consideration, what is best for the design of the plant and future regulatory interactions. However, for purposes of completing the LMP process, two SSC candidate options for being classified as safety-related are carried forward to illustrate the corresponding DBAs.

3.5.2 Safety-Related SSCs for Remove Core Heat Example Safety Function

"Remove Core Heat" (RCH) is a required safety function for any nuclear reactor with a small but non-zero heat generation (i.e., residual heat). The time available to perform this required safety function may vary between reactor designs. Additionally, the criteria for when this safety function is sufficiently satisfied varies between reactor designs. For the HTGC-PBR Demonstration, given the low core power density and high heat capacity, the geometry of the reactor vessel, RB, and surrounding heat sinks are sufficient to meet these criteria without any active systems, with or without pressurized helium. This is illustrated in Figure 9, which shows an additional level of core heat removal required sub-functions, which includes transfer core heat to the reactor vessel, radiate heat from the reactor vessel wall, transfer heat to ultimate heat sinks, and maintain geometry for conduction and radiation.

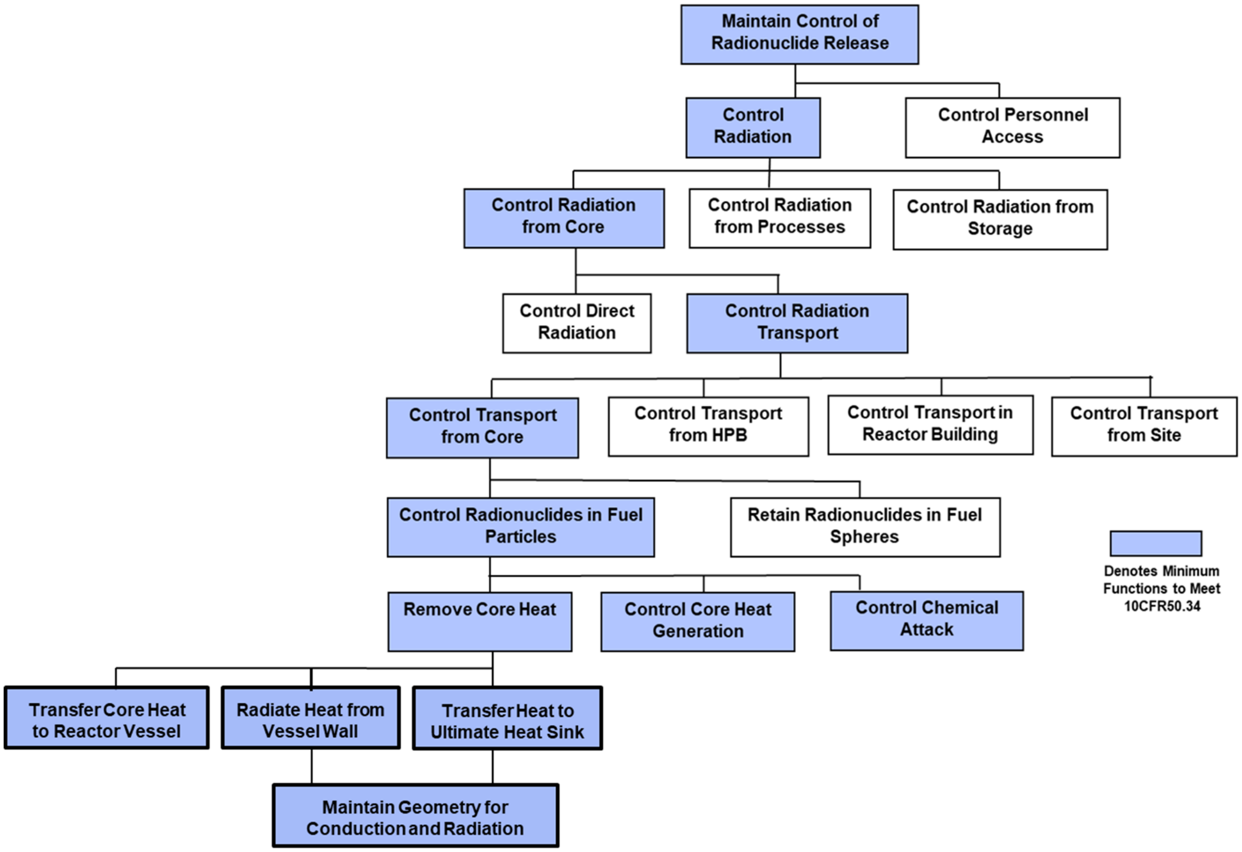

Figure 9. Core Heat Removal Required Sub-Functions for the HTGC-PBR Demonstration

Returning to the Small Helium Depressurization (SD) event sequence, Figure 3 shows three sets of SSCs that are available to remove core heat:

- The main loop that circulates the heat from the core to the Steam Generator where it is transferred to heat sinks in the conventional steam cycle
- The startup/shutdown heat removal system, which takes the residual core heat from the Steam Generator to diverse heat sinks in the nuclear island
- The passive heat removal by heat sinks surrounding the reactor vessel, such as the RCCS, which also has an active mode

Table 3 examines the core heat removal function for the small helium depressurization DBEs to see if these or other sets of SSC choices are available. For any DBE with a set of SSCs that are not available, that set of SSCs is then not an option for performing the required safety function. For the small depressurization DBEs, only the last two passive safety SSC sets are viable options for performing the required safety function for all DBEs. This confirms that these are required safety functions as indicated in Figure 9.

Table 3. Sets of SSCs Available During Small Helium Depressurization DBEs

Alternate Sets of SCCs	Small Helium Depressurizations			Option for Required Safety Function?
	DBE-5	DBE-10	DBE-11	
• Reactor Core • Steam Generator + Circulator • ML Forced Cooling	N	Y	N	No
• Reactor Core • Steam Generator + Circulator • SU/SD - Active	Y	Y	N	No
• Reactor Core • Reactor Vessel • RCCS - Active	Y	Y	N	No
• Reactor Core • Reactor Vessel • RCCS - Passive	Y	Y	Y	Yes
• Reactor Core • Reactor Vessel • RB/Ground + Air Heat Sinks	Y	Y	Y	Yes

In Table 4, all 17 DBEs from all 11 IEs are examined together for core heat removal. By including events that involve a loss of the active mode of RCCS power, only the last two rows of Table 3 contain sets of SSCs that satisfy the required safety function.

Table 4. Available Candidate SSCs for Required Safety Function to Remove Core Heat over DBE

Alternate Sets of SCCs	Small Helium Depressurizations																	Option for Required Safety Function?
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	
• Reactor Core • Steam Generator + Circulation • ML Forced Cooling	N	Y	N	N	N	N	N	N	N	Y	N	N	N	N	N	N	N	No
• Reactor Core • Steam Generator + Circulation • SU/SD - Active	Y	Y	Y	Y	Y	N	N	Y	Y	Y	N	N	Y	N	N	N	Y	No
• Reactor Core • Reactor Vessel • RCCS - Active	Y	Y	Y	Y	Y	N	N	Y	Y	Y	N	N	Y	Y	Y	Y	Y	No
• Reactor Core • Reactor Vessel • RCCS - Passive	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Yes
• Reactor Core • Reactor Vessel • RB/Ground + Air Heat Sinks	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Yes

The passive heat sinks have a high reliability and are available for DBEs with extended Loss of Offsite and Onsite power. Therefore, as indicated in Table 4, only two options remain viable to select for being safety-related for the Remove Core Heat function. It should be noted that the

first three SSC sets might be viable options for being classified as safety-related if their system reliabilities were enhanced such that upon re-evaluation of the LBEs, no system failures would be observed to occur for events in the DBE range.

The designer has many design requirements to balance including performance, economics, market, and time-to-market. Having multiple choices to achieve safety requirement is an advantage. There are many approaches to weighing choices to best meet requirements with the least risk. Table 5 provides a simplified example for the choice of which set of SSCs to classify as safety-related for the RCH required safety function.

Table 5. Relative Performance, Cost, and Risk of Available Candidate SSC Choices for the Required Safety Function "Remove Core Heat"

SSC Candidate Options		Capability	Reliability	Design, Dev., and Capital Cost	FSAR/Regulatory Special Treatment Cost
RCH-1	• Reactor Core • Reactor Vessel • RCCS - Passive	High	High	Medium	Low
RCH-2	• Reactor Core • Reactor Vessel • RB and Ground + Air Heat Sinks	High	High	High	High

In the first two performance columns of capability and reliability, where a high rating is desirable, there is not much differentiation among the choices. However, in the cost and risk columns, the last choice of relying on the reactor building and the surrounding ground and air as the passive heat sinks, which upon first impression is the simplest and easiest to understand, may lead to specialized materials, sites, analyses, and costly confirmatory testing.

3.5.3 Safety-Related SSCs for Control Heat Generation Example Safety Function

Like the required RCH safety function, this safety function is common to any commercial nuclear reactor. Power increases are only desired to occur by operator intervention. Likewise, power decreases are desired when any system perturbation occurs which could lead to an undesirable plant end state. One negative feedback mechanism common to all thermal-spectrum based reactors is the fuel temperature. As fuel temperature increases, power tends to decrease as more neutrons are captured without initiating a fission event. This inherent property of the HTGC-PBR is identified as part of the sets of the SSCs which can satisfy the function "Control Heat Generation" (CHG).

Table 6 provides a visual representation of the process for identifying SSC sets which can perform the required CHG safety function. In Table 6, the Reactor refers to the core, including the reflectors. The Operating Control System and Operating Control Rods are needed to control the nuclear reaction during normal operation. Similarly, the Investment Protection System (IPS), Reactor Protection System (RPS) and Shutdown Control Rods can also be used to shut down the nuclear reaction. The motivation for this system is to have an additional redundant, and diverse

system for reactivity control. If available, either of these systems can satisfy the CHG required safety function. The third SSC set option does not include either control rod system. This relies strictly on the passive, negative feedback mechanism associated with fuel heat up.

Table 6. Available Candidate SSCs for Required Safety Function "Control Heat Generation"

Alternate Sets of SCCs	Spectrum of Design Basis Events																	Option for Required Safety Function?
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	
• Reactor • Operating Control System/Operating Control Rods • Fuel Temp. Coefficient	Y	Y	Y	Y	Y	Y	Y	N	Y	Y	Y	Y	Y	Y	Y	Y	Y	No
• Reactor • IPS/RPS/Shutdown Control Rods • Fuel Temp. Coefficient	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Yes
• Reactor • Fuel Temp. Coefficient	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Yes

Since DBE-8 involves an inadvertent control rod withdrawal IE, this SSC is not available. Thus, the first set of SSCs is not a viable option for being safety-related.

3.5.4 Safety-Related SSCs for Control Water Chemical Attack Required Safety Function

"Control Water Chemical Attack" is a required safety function which is a part of the larger required safety function of "Control Chemical Attack," as shown in Figure 7. The "Control Chemical Attack" function also considers air-oxidation which would lead to the required safety function "Control Air Chemical Attack." However, for the purposes of the HTGR-PBR Demonstration, only water chemical attack is considered. Water chemical attack is a hazard for any HTGR which contains a water/steam source coupled to the primary loop, such as a steam generator, as is the case for the HTGC-PBR.

Table 6 provides a visual representation of the process for identifying SSC sets which can perform the required safety function "Control Water Chemical Attack." The first set of SSCs include the Reactor Core and the two Steam Generator tube bundles. The Reactor Core is a critical SSC since it provides a significant source of non-fuel element graphite for oxidation which would be sacrificed prior to fuel element oxidation. The SG tube bundles are the normal operation SSC which prevents water ingress and chemical attack. However, for DBE-1, this is a Steam Generator Tube Rupture IE, so this SSC set is not a viable option for safety-related classification.

Table 7. Available Candidate SSCs for Required Safety Function "Control Water Chemical Attack"

Alternate Sets of SCCs	Spectrum of Design Basis Events																	Option for Required Safety Function?
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	
• Reactor Core • SG Tube Bundles	N	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	No
• Reactor Core • IPS/Moisture Monitor • Circ. Trip/Feedwater-Main Steam Isolation	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Yes
• Reactor Core • RPS Helium Pres. and Fluence • Circ. Trip/Feedwater-Main Steam Isolation	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Yes
• Reactor Core • RPS Helium Pres. and Fluence • Circ. Trip/Feedwater-Main Steam Isolation • SG Dump	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Yes

The other SSC sets involve a method for indicating a tube rupture event, and then a method to halt the water ingress or limit the consequences. For each safety function, analyses and/or experimental data will be needed to verify the success or satisfaction adequacy of the required safety functions by the selected SSCs.

3.5.5 Preview of DBA Identification for Two Example Safety-Related SSC Sets

Two example safety-related SSC sets are identified for the purposes of identifying example corresponding DBAs. These examples are shown in Table 8.

Table 8. Two Example Safety-Related SSC Sets Based on the LMP Process for the HTGC-PBR

Option 1	Remove Core Heat	Control Heat Generation
Option RCH-1 CHG-1	• Reactor Core • Reactor Vessel • RCCS - Passive	• Reactor Vessel + Reactor Core • Control Rods • Fuel Temperature Coefficient
Option 2	**Remove Core Heat**	**Control Heat Generation**
Option RCH-2 CHG-2	• Reactor Core • Reactor Vessel • RB and Ground + Air Heat Sinks	• Reactor Core • Fuel Temperature Coefficient

For each option shown in Table 8, and for each of the 17 DBEs listed in Table 2, a corresponding DBA is identified, which is shown in Figure 10. For the purposes of this HTGC-PBR Demonstration, only one DBA is illustrated for each of the two SSC options. This is the small depressurization event DBE-5 discussed above with successful forced cooling via the alternate

SU/SD system. DBE-5 is shown in blue with the entire plant responding, whereas, the DBA options only have the safety-related SSCs successfully responding. DBA Option 1 shown in red has successful reactor trip with control rods and core heat removal with the passive RCCS. DBE Option 2 in green controls heat generation with the negative temperature coefficient alone and core heat removal with the reactor building and surroundings.

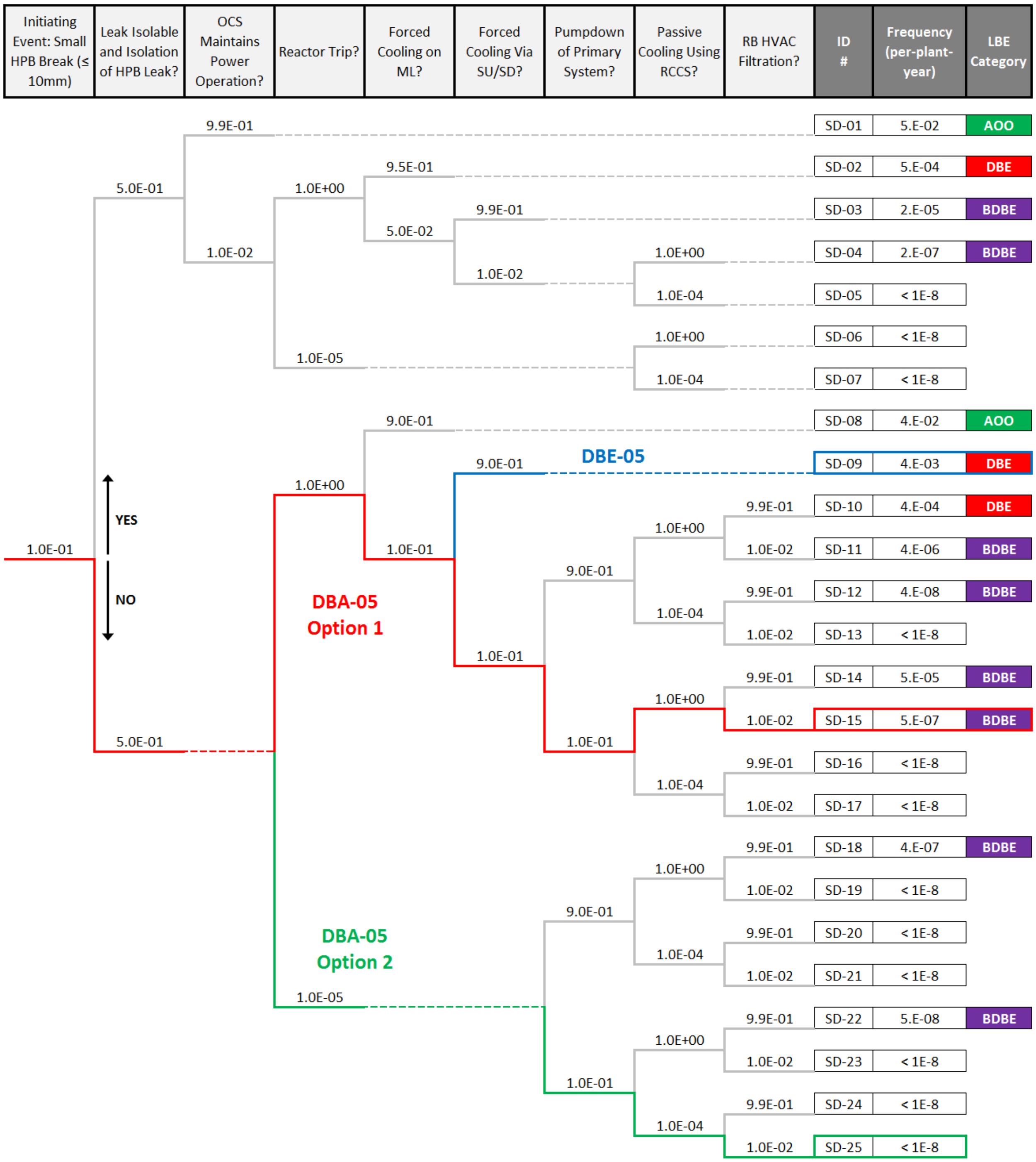

Figure 10. DBE-5 and Corresponding DBA-5 Options with Only Safety-Related SSCs Operable as Illustrated on Small Helium Depressurization Event Tree

3.5.6 LMP Process Feedback for SSC Safety Classification

The LMP process presents a systematic and reproducible process for SSC classification as shown in the Demonstration with Xe-100 examples. One of the major takeaways from the Demonstration is recognizing the importance of involving the key decision makers on the design team during all phases of the LMP proposed SSC safety classification process. For example, the designers identified several additional sets of SSCs that could be classified as safety-related within the LMP process.

3.6 Introduction to the LMP Process for Defense-in-Depth Adequacy Evaluation

The final key part of the LMP RIPB framework addressed in the Demonstration is the RIPB evaluation of DID adequacy. The reasons for including a DID evaluation as part of the LMP framework are discussed below.

- It is desirable for the reactor developer and the plant operator to take ownership of establishing DID adequacy because the key RIPB decisions that influence plant capabilities and programs that are responsible for achieving DID adequacy must be introduced early in the design.
- Historical references on DID by the NRC[12] and the International Atomic Energy Agency (IAEA)[13] are rooted in terms that were developed for operating LWRs and have questionable applicability to advanced non-LWRs. Hence a technology inclusive language for defining and evaluating DID is needed.
- By documenting the basis for DID adequacy as part of the safety case it is considered less likely that there would be costly back fits imposed by the regulator and more likely that the regulator would appreciate the DID capabilities.
- The LMP approach to DID includes criteria for deciding the sufficiency of DID that aim to avoid an open-ended process for evaluating DID. The approach is based on the Layers of Defense concept recommended in Reference [13] which acknowledges both physical barriers and functional barriers to prevent the uncontrolled release of radioactive material.

An overview of the key elements of the approach to defining DID, which is built on earlier efforts to address this topic for the PBMR and NGNP projects, is provided in Figure 11. The LMP approach to establishing DID adequacy is integrated into each of the other key elements of the LMP framework. There are important DID roles in each major element of the framework including:

- Designer development of the safety design approach
- Development and analysis of information from the design and design specific PRA development
- Selection and evaluation of LBEs
- Establishing the adequacy of margins in the evaluation of risk significance of LBEs, safety functions, and SSCs

- SSC safety classification and development of SSC performance requirements
- Establishing the appropriate special treatment based on the insights gained from the PRA LBE development and SSC classification

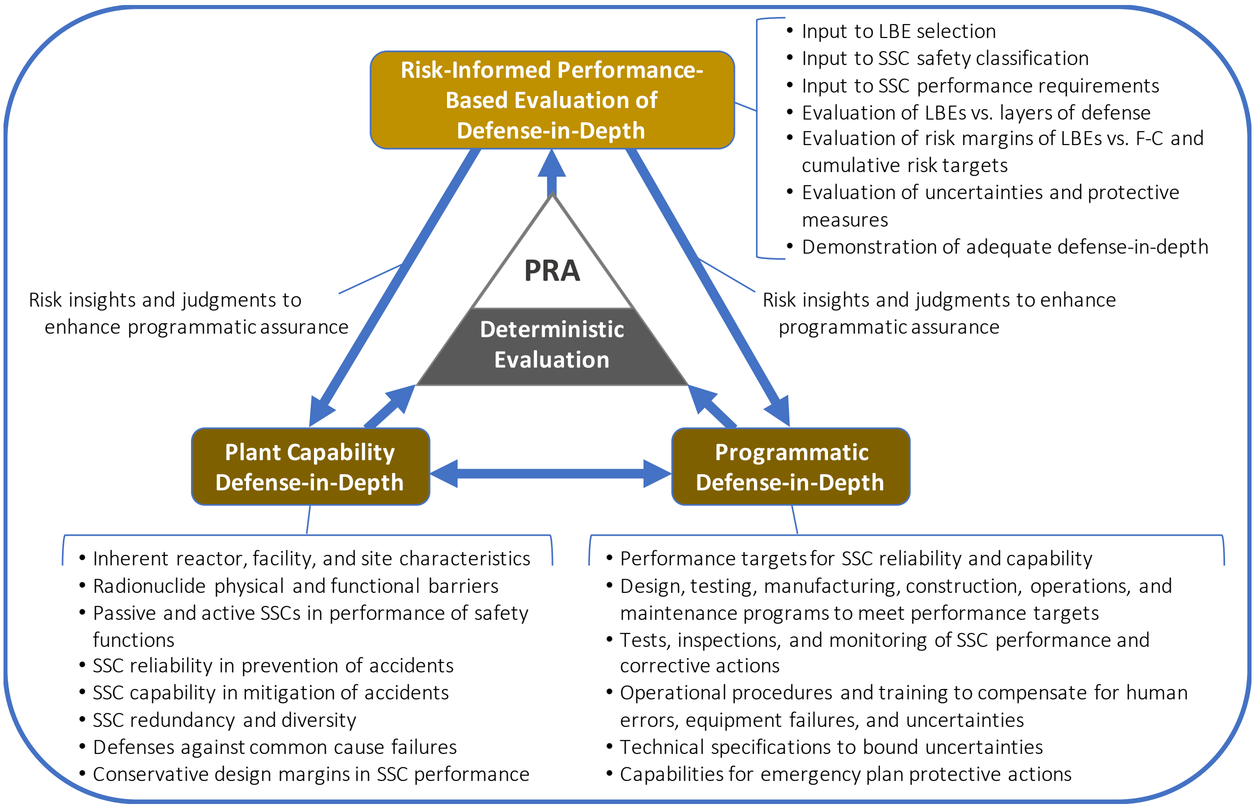

Figure 11. LMP Framework for Establishing DID Adequacy

To address the key question of "When is enough, enough?" a set of criteria are included as summarized in Table 9 for application in establishing DID adequacy for a plant at a suitable level of design completion. Given the limitations of the Rev. 0 PRA and current level of design completion for the X-energy reactor, assessment of DID for such reactor was not completed as part of the scope of this Demonstration. When implemented, these criteria track the five key layers of defense in the prevention and mitigation of accidents, and include both quantitative and qualitative criteria for establishing DID adequacy. These criteria ensure that the frequencies and consequences of LBEs are maintained in the appropriate categories and exhibit sufficient margins against the F-C target and the cumulative risk criteria. The adequacy is evaluated at each stage of design, licensing, construction, and operation by an Integrated Decision Panel, comprised initially of those responsible for selecting and evaluating the safety design approach and eventually by the owner operator. These DID evaluations are documented and available for review and audit by the regulator.

Table 9. Guidelines for Establishing the Adequacy of Overall Plant Capability Defense-in-Depth

<table>
<tr><th rowspan="2">Layer[a]</th><th colspan="2">Layer Guideline</th><th colspan="2">Overall Guidelines</th></tr>
<tr><th>Quantitative</th><th>Qualitative</th><th>Quantitative</th><th>Qualitative</th></tr>
<tr><td>1) Prevent off-normal operation and AOOs</td><td colspan="2">Maintain frequency of plant transients within designed cycles; meet user requirements for plant reliability and availability[b]</td><td rowspan="5">Meet F-C target for all LBEs and cumulative risk metric targets with sufficient[d] margins</td><td rowspan="5">No single design or operational feature, [c] no matter how robust, is exclusively relied upon to satisfy the five layers of defense</td></tr>
<tr><td>2) Control abnormal operation, detect failures, and prevent DBEs</td><td>Maintain frequency of all DBEs < 10^{-2}/ plant-year</td><td>Minimize frequency of challenges to safety-related SSCs</td></tr>
<tr><td>3) Control DBEs within the analyzed design basis conditions and prevent BDBEs</td><td>Maintain frequency of all BDBEs < 10^{-4}/ plant-year</td><td>No single design or operational feature[c] relied upon to meet quantitative objective for all DBEs</td></tr>
<tr><td>4) Control severe plant conditions, mitigate consequences of BDBEs</td><td rowspan="2">Maintain individual risks from all LBEs < Quantitative Health Objectives (QHOs) with sufficient[d] margins</td><td rowspan="2">No single barrier[c] or plant feature relied upon to limit releases in achieving quantitative objectives for all BDBEs</td></tr>
<tr><td>5) Deploy adequate offsite protective actions and prevent adverse impact on public health and safety</td></tr>
</table>

Notes:

[a] The plant design and operational features and protective strategies employed to support each layer should be functionally independent.

[b] Non-regulatory user requirements for plant reliability and availability and design targets for transient cycles should limit the frequency of initiating events and transients and thereby contribute to the protective strategies for this layer of DID. Quantitative and qualitative targets for these parameters are design specific.

[c] This criterion implies no excessive reliance on programmatic activities or human actions and that at least two independent means are provided to meet this objective.

[d] The level of margins between the LBE risks and the QHOs provides objective evidence of the plant capabilities for DID. Sufficiency will be decided by the designer's Integrated Decision Panel (IDP).

4.0 CONCLUSIONS AND OBSERVATIONS

4.1 Conclusions

The Demonstration Project met the objectives and deliverables as outlined in the project charter and summarized in this report.

Objective 1 Conclusions

Objective 1: Demonstrate key processes within the LMP Guidance Document to X-energy.

Significant progress was made to demonstrate the selection of LBEs based on information obtained from the Xe-100 Phase 0 PRA for event sequences, combined with performance-based targets for frequency and radiological dose, reflecting the Xe-100 pre-conceptual design and additional efforts to estimate offsite radiological doses for each LBE. The process of defining the required safety functions using Xe-100 examples was also demonstrated. Options for selecting safety-related SSCs for each required safety function were identified in these examples. However, it should be noted that classification of SSCs as safety-related will ultimately be made by the designer. An introduction to the process of evaluating DID adequacy was also reviewed.

Based on the scope and design basis of this Demonstration Project, the process described in the LMP Guidance Document was shown to provide a systematic and reproducible framework for selection of LBEs, defining required safety function, and classification of SSCs. However, The Guidance Document lacked detail on the topic of safety functions and their linchpin relationship between the selection of LBEs and the Safety Classification of SSCs. As a result of this Demonstration, a refinement to the LBE selection flow chart was recommended and has been incorporated into the latest draft of the Guidance Document to provide more visibility to the role of required safety functions.

Objective 2 Conclusions

Objective 2: Leverage the LMP process to improve the regulatory certainty of X-energy's Xe-100 design and safety case, as best possible at the current state of design, by identifying a credible suite of Licensing Basis Events and investigating available structure, system, and component groupings that result in acceptable outcomes for the identified LBEs. Underlying this objective is the assertion that use of risk informed, performance based methods to reach these conclusions are endorsed by Commission policy and compatible with the existing regulatory framework.

The Demonstration provided insights into how the technology-inclusive, RIPB LMP process can be incorporated during the pre-conceptual design and subsequent phases of the Xe-100 by providing a process to reduce regulatory uncertainty; specifically, in the areas of LBE selection, identification of required safety functions, SSC classification, and DID adequacy.

It is expected that the LMP Demonstration Project, and this associated report, will serve as an input into X-energy's regulatory engagement strategy. Additionally, some important safety design questions were raised as part of the Demonstration. While no decisions were made during this Demonstration, an understanding of the RIPB processes proved valuable in identifying the

steps needed to evaluate the safety-risk associated with various design selections and the tradeoffs needed from other stakeholders to achieve broader designer objectives.

One example of a design selection is the selection of safety related SSC sets. An initially selected safety related SSC may change after the LMP process is performed given some newly identified options for satisfying the required safety functions. This selection then becomes a design requirement on those new or modified SSCs. Those SSCs must have a certain design maturity, based on a list of initial functions and requirements, to base, at least, a preliminary-level PRA, like the preconceptual Xe-100 Phase 0 PRA, on. This process highlighted some iterative design steps which may occur as a result of performing the LMP processes.

In summary, this Demonstration shows the LMP processes have the potential to reduce regulatory uncertainty by providing a systematic and transparent process, as well as criteria, for addressing key and fundamental safety questions and informing design decisions.

4.2 Observations

- For the LMP to be fully effective, organizations should provide internal training on the LMP process, and ensure the appropriate members are sufficiently involved.
    - Decision makers beyond the core engineering staff should use/embrace and be sufficiently knowledgeable of the LMP process to ensure regulatory/licensing success.
    - It is important for the licensing team following the LMP approach to have key engineering design team members and decision makers in charge of the safety function development and the selection of SSC safety classification.
- LMP provides a RIPB framework for making decisions that impact the safety design approach and technical basis for licensing activities. Based on the Demonstration, X-energy feels like the usefulness and benefits of the LMP RIPB process will increase over time as the design completion - PRA update integration processes are performed.
- The LMP approach to functional design criteria complements more deterministic approaches to Principle Design Criteria. Additional effort is needed to define the relationship of the LMP process with existing reactor General Design Criteria, modular HTGR Advanced Reactor Design Criteria, and other licensing requirements. These are topics currently being discussed as the Guidance Document and associated NRC interactions move forward.
- RIPB decisions in the LMP framework are made by the designer (e.g. by an IDP, or equivalent) for approval by the regulator. This exercise of the LMP framework initially demonstrated that the RIPB processes can successfully:
    - Provides a process for selecting and evaluating LBEs including the selection of DBAs as defined by user selected safety-related SSCs for the performance of required safety functions.
    - Provides criteria for establishing risk significant and safety significant SSCs. Based on insights from the Xe-100 Demonstration and previous HTGR examples, it is likely

that for X-energy's Xe-100, most risk significant SSCs will be contained within the set of safety-related SSCs because none of the SSCs beyond the SR SSCs are expected to meet the LMPs criteria for risk significance.

  - Enables the designer to establish requirements for the SSC reliabilities and capabilities to prevent and mitigate LBEs that will flow down to the special treatment requirements for safety significant (SR and NSRST SSCs).
  - Gives the designer ownership of the responsibility to evaluate the adequacy of DID whose documentation will be available for review and audit by the NRC.
  - The Guidance Document provides flexibility to the designer to meet other top-level requirements (e.g. performance, cost, risk, etc.).

- Once DID adequacy evaluations are finalized, reliability and performance requirements (e.g. technical specifications, reliability assurance programs, limits of operation, surveillance requirements, etc.) can be constructed for all the SR and any NSRST SSCs. Margin to the F-C target can lead to the derivation of less stringent performance requirements than what has been typically done for existing light water reactors.
- There was some discussion of the LMP criteria for risk significant LBEs. For a given level of consequence, LBEs with frequencies within 1% of the F-C target frequency are regarded as risk significant. The basis for this is the ASME/ANS PRA Standard RA-S-1.4-2013.

5.0 REFERENCES

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[4] PBMR (Pty) Ltd., "US Design Certification, Licensing Basis Event Selection for the Pebble Bed Modular Reactor," ADAMS Accession Number ML061930123, June 2006.

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[9] 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," Draft, ADAMS Accession Number ML17290A463, October 2017.

[10] 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," Draft, ADAMS Accession Number ML17354B174, December 2017.

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[13] International Atomic Energy Agency, Safety Report Series No. 46, "Assessment of Defense in Depth for Nuclear Power Plants," 2005.

From: Bolin, John
To: Reckley, William
Subject: [External_Sender] X-energy LMP Demonstration Project Report
Date: Friday, August 10, 2018 12:22:33 PM
Attachments: X-energy LMP Demonstration Project Report final 2018-08-08.pdf

Hi Bill,

As the LMP's tabletop exercises overall lead, I am happy to share with you and the NRC advanced reactor staff the attached final report on a tabletop exercise with X-energy on their advanced reactor design. This is the first of at least three other tabletop reports that will be generated. Work is progressing on tabletop exercises for the GE PRISM and an MSRE design and additional vendor teams may also participate. I have separately sent this report to Kati Austgen for the use of the NEI Advanced Reactor Regulatory Task Force.

Thanks for all of your efforts
John Bolin

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