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

Selection of Licensing Basis Events
Draft Report Revision 0

Prepared for the
U.S. Department of Energy
Office of Nuclear Energy
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

April 2017

ABSTRACT

This report, Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, Licensing Basis Event (LBE) Selection, represents a key element in the development of a framework for the efficient licensing of advanced non-light water reactors (non-LWRs). It is the result of a Licensing Modernization Project (LMP) led by Southern Company and cost-shared by the United States Department of Energy (DOE). The LMP will result in detailed proposals for establishing licensing technical requirements to facilitate risk-informed and performance-based design and licensing of advanced non-LWRs. Such a framework acknowledges enhancements in safety achievable with advanced designs and reflects more recent states of knowledge regarding safety and design innovation, creating an opportunity for reduced regulatory complexity with increased levels of safety. The project builds on best practices as well as previous activities through DOE and industry-sponsored advanced reactor licensing initiatives.

The LMP objective is to assist the NRC to develop regulatory guidance for licensing advanced non-LWR plants.

This paper presents a modern, technology-inclusive, risk-informed, and performance-based (TI-RIPB) approach to identifying LBEs, which cover a spectrum of events considered in the design and licensing of a nuclear power plant. A key licensing outcome of this approach is the structured selection of design basis accidents (DBAs) that are traditionally analyzed in Chapter 15 of the license application. In this paper, the LMP is seeking:

(1) NRC's approval of the proposed LBE selection approach for incorporation into appropriate regulatory guidance;
(2) Identification of any issues that have the potential to significantly impact the selection and evaluation of LBEs, including anticipated operational occurrences (AOOs), design basis events (DBEs), beyond design basis events (BDBEs) and design basis accidents (DBAs)

Development of the LBE selection approach begins with a review of the relevant regulatory policy and available guidance for selecting LBEs. From this review desirable attributes of an LBE selection and evaluation process are defined and used to develop the proposed approach. This paper describes the methodology for selecting and evaluating LBEs, and sets forth issues for resolution in order to facilitate an effective submittal leading to license applications for advanced non-LWRs.

This paper builds on the development and subsequent NRC staff and ACRS reviews of an LBE white paper for DOE's Next Generation Nuclear Plant (NGNP), a modular high-temperature gas-cooled reactor (HTGR), which was derived from earlier precedents on the MHTGR and PBMR. The proposed LBE method is intended for use with the full spectrum of advanced non-LWR concepts currently under consideration for development. The technology-inclusive capabilities of the proposed method are demonstrated using example LBEs from the MHTGR and PRISM. The information in this paper is intended to serve as the basis for interactions with the NRC staff leading to the development of regulatory guidance for the preparation of license applications.

Executive Summary

INTRODUCTION

This report represents a key element in the development of a framework for the efficient licensing of advanced non-light water reactors (non-LWRs). It is the result of a project led by Southern Company and cost-shared by the United States Department of Energy (DOE). This Licensing Modernization Project (LMP) will result in detailed proposals for establishing licensing technical requirements to facilitate efficient design and licensing of advanced non-LWRs. This paper presents a modern, technology-inclusive, risk-informed, and performance-based (TI-RIPB) approach to identifying a full set of licensing basis events (LBEs) used in the design and licensing of advanced non-LWRs. A key licensing outcome of this process is the structured, systematic, and reproducible process for the selection of design basis accidents (DBAs) for advanced non-LWR plants. Additional LMP papers are planned to address other RIPB decisions within the licensing framework.

DEFINITION OF LICENSING BASIS EVENTS (LBEs)

As the term is used in this document, LBEs are defined broadly to include all the events used to support the safety aspects of the design[1] and to meet licensing requirements. They cover a comprehensive spectrum of events from normal operation to rare, off-normal events. There are four categories of LBEs:

- Anticipated Operational Occurrences (AOOs), which encompass planned and anticipated events whose frequencies exceed 10^{-2}/plant-year where a plant may be comprised of one or more reactor modules. The radiological doses from AOOs are required to meet normal operation public dose requirements. AOOs are utilized to set operating limits for normal operation modes and states.
- Design Basis Events (DBEs) encompass unplanned off-normal events not expected in the plant's lifetime whose frequencies are in the range of 10^{-4} to 10^{-2}/plant-year, but which might occur in the lifetimes of a fleet of plants. DBEs are the basis for the design, construction, and operation of the structures, systems, and components (SSCs) during accidents and are used to provide input to the definition of design basis accidents (DBAs).
- Beyond Design Basis Events (BDBEs), which are rare off-normal events whose frequencies range from $5x10^{-7}$/plant-year to 10^{-4}/plant-year. BDBEs are evaluated to ensure that they do not pose an unacceptable risk to the public.
- Design Basis Accidents (DBAs). The DBAs for Chapter 15, "Accident Analyses," of the license application are prescriptively derived from the DBEs by assuming that only SSCs classified as safety-related are available to mitigate the consequences. The public consequences of DBAs are based on mechanistic source terms and are conservatively evaluated.

LMP APPROACH TO SELECTING AND EVALUATING LBEs

The technology-inclusive, risk-informed, and performance-based (TI-RIPB) approach to selecting LBEs is designed to ensure that an appropriate set of limiting events for each reactor technology are reflected in the selection of DBAs and that the full set of LBEs define the risk significant events for each design and technology. This is essential to ensure that risk insights are appropriately reflected in the design and licensing decisions including the selection of DBAs.

---

1 LBEs do not cover all the events used to support the design, only those to meet safety requirements. There are other events considered in the design that do not necessarily impact safety performance but are used to ensure protection of the investment and to meet plant reliability, availability, and capacity factor targets.

The LBEs in each category are evaluated individually to support the tasks of assessing the performance of SSCs with respect to safety functions in response to initiating events and collectively to demonstrate that the integrated risk of a multi-reactor module plant design meets the NRC Safety Goals. An important outcome of the selection and evaluation of LBEs is to identify design features of the plant that are necessary and sufficient to ensure that risk goals are achieved and licensing requirements are met. The use of these insights in the derivation of performance requirements and principle design criteria for SSCs, including the radionuclide barriers, is a topic of a future LMP white paper on SSC safety classification. The key licensing outcome is the systematic derivation of the DBAs.

On the basis of the lessons learned from a regulatory precedent review that is described in this paper and the objectives of the LMP, the process for selecting LBEs for advanced non-LWRs should be:

- **Systematic and Reproducible**
- **Sufficiently Complete**
- **Available for Timely Input to Design Decisions**
- **Risk-informed and Performance-Based:**
- **Reactor Technology Inclusive**
- **Consistent with Applicable Regulatory Requirements**

A flow chart indicating the steps to identify and evaluate LBEs in concert with the design evolution is shown in Figure ES-1. These steps are intended to be carried out by the design and design evaluation teams responsible for establishing the key elements of the safety case and preparing a license application. The process is used to prepare an appropriate licensing document, e.g., licensing topical report, that documents the derivation of the LBEs, which would be reviewed by the regulator as part of license review. The design and design evaluation teams are responsible for selecting the LBEs and justifying their selections. The regulator is responsible to review the design, the LBE selections, and their derivation. Although the NRC is expected to review the entire LBE selection and evaluation process, the specific steps with increased regulatory involvement are indicated in the attached figure.

The process is implemented in the following LBE selection tasks:

Task 1 Propose Initial List of LBEs
In order to begin the design, it is necessary to select an initial set of LBEs which may not be complete but is necessary to develop the basic elements of the safety design approach. These events are selected deterministically based on all relevant and available experience including experience from the design and licensing of reactors of a different technology.

Task 2 Design Development and Analysis
The design development is performed in phases and often includes pre-conceptual, conceptual, preliminary, and final design phases and may include iterations within phases. The subsequent Tasks 3 through 9 are repeated for each design phase until the list of LBEs is finalized.

Task 3 PRA Development/Update
A PRA model is developed and updated for each phase of the design. In the first design phase, which is typically the pre-conceptual design, the PRA is of limited scope and coarse level of detail and makes use of engineering judgment much more than a completed PRA that would meet applicable PRA standards. The scope and level of detail of the PRA are then enhanced as the design matures and siting information is defined.

Task 4 Identify/Revise List of AOOs, DBEs, and BDBEs
The event sequences modeled and evaluated in the PRA are grouped into accident families each having a similar initiating event, challenge to the plant safety functions, plant response, and mechanistic source term if there is a release.

Task 5 Select/Revise Safety-Related SSCs
Tasks 5 and 6 are performed together rather than sequentially. In Task 6 all the DBEs are subject to a prescriptive evaluation that involves the determination of which safety functions are necessary and sufficient to ensure that 10 CFR 50.34 dose requirements can be met based on a conservative analysis for each safety function challenge represented in each DBE. In Task 5 the design team makes a decision on which SSCs that perform these required safety functions should be classified as safety related for each DBE.

Figure ES-1 Process for Selecting and Evaluating Licensing Basis Events

Task 6 Select DBAs

For each DBE identified in Task 4, a DBA is defined that includes the required safety function challenges represented in the DBE, but assumes that the required safety functions are performed exclusively by safety-related SSCs. These DBAs are then used in Chapter 15 of the license application for supporting the conservative deterministic safety analysis.

Task 7 Perform LBE Evaluations

The deterministic and probabilistic safety evaluations that are performed for the full set of LBEs are covered in the following five tasks:

Task 7a. Evaluate LBEs against TLRC Frequency – Dose Criteria

In this task the results of the PRA which have been organized into LBEs will be evaluated against the TLRC frequency-consequence criteria of Figure ES-2. The evaluations in this step are performed on each LBE separately. The mean values and the uncertainties associated with those means are used to classify the LBEs into AOOs, DBEs, and BDBE categories. Part of the LBE frequency-dose evaluation is to ensure that LBEs involving releases from two or more reactor modules do not make a significant contribution to risk and to ensure that measures to manage the risks of multi-module accidents are taken to keep multi-module releases out of the list of DBAs. Another key element of this step is to identify design features that are responsible for meeting the frequency-dose criteria including those that are responsible for preventing any release for those LBEs where applicable. This evaluation leads to performance requirements and design criteria that are developed within the framework of the SSC classification step in the TI-RIPB design and licensing approach.

Task 7b. Evaluate Integrated Plant Risk

In this task, the integrated risk of the entire plant is evaluated against four criteria as follows:

- The total frequency of exceeding an offsite boundary dose of 100 mrem shall not exceed 1/plant-year to ensure that the annual exposure limits in 10 CFR 20 are not exceeded.
- The total frequency of an offsite boundary dose exceeding 750 rem shall not exceed 10^{-6}/plant-year. Meeting this criterion satisfies the NRC Safety Goal Policy Statement on limiting the frequency of a large release.
- The average individual risk of early fatality within the area 1 mile of the EAB shall not exceed $5x10^{-7}$/plant-year to ensure that the NRC Safety Goal Quantitative Health Objective (QHO) for early fatality risk is met
- The average individual risk of latent cancer fatalities within the area 10 miles of the EAB shall not exceed $2x10^{-6}$/plant-year to ensure that the NRC safety goal QHO for latent cancer fatality risk is met.

Another key element of this step is to identify design features that are responsible for meeting the integrated risk criteria. This evaluation leads to performance requirements and design criteria that are developed within the framework of the SSC classification step in the TI-RIPB design and licensing approach.

Task 7c. Evaluate risk significance of Barriers and SSCs

In this task, the details of the definition and quantification of each of the LBEs in Task 7a and the integrated risk evaluations of Task 7b are used to define both the absolute and relative risk significance of individual SSCs and radionuclide barriers. These evaluations employ technology inclusive risk importance metrics and an examination of the effectiveness of each of the barriers in retaining radionuclides. This information is used to provide risk insights to the design team and to support the RI-PB evaluation of defense-in-depth in Task 7e.

Task 7d. Perform Deterministic Safety Analyses against 10 CFR 50.34
This task corresponds to the traditional deterministic safety analysis that is found in Chapter 15 of the license application. It is performed using conservative assumptions. The uncertainty analyses in the mechanistic source terms and radiological doses that are part of the PRA are available to inform the conservative assumptions used in this analysis and to avoid the arbitrary "stacking" of conservative assumptions.

Task 7e. Risk-Informed, Performance-Based Evaluation of Defense-in-Depth
In this task, the definition and evaluation of LBEs will be used to support a RI-PB evaluation of defense-in-depth. This task involves the identification of key sources of uncertainty, characterization of safety margins, and evaluation against defense-in-depth criteria that are the subject of a companion white paper to be developed in the LMP as a future deliverable.

Task 8 Decide on Completion of Design/LBE Development
The purpose of this task is to make a decision as to whether additional design development is needed to select the LBEs, either to proceed to the next logical stage of design or to incorporate feedback from the LBE evaluation that design improvements should be considered. Such design improvements could be motivated by a desire to increase margins against the frequency-consequence criteria, reduce uncertainties in the LBE frequencies or consequences, manage the risks of multi-reactor-module accidents, or enhance the performance against defense-in-depth criteria.

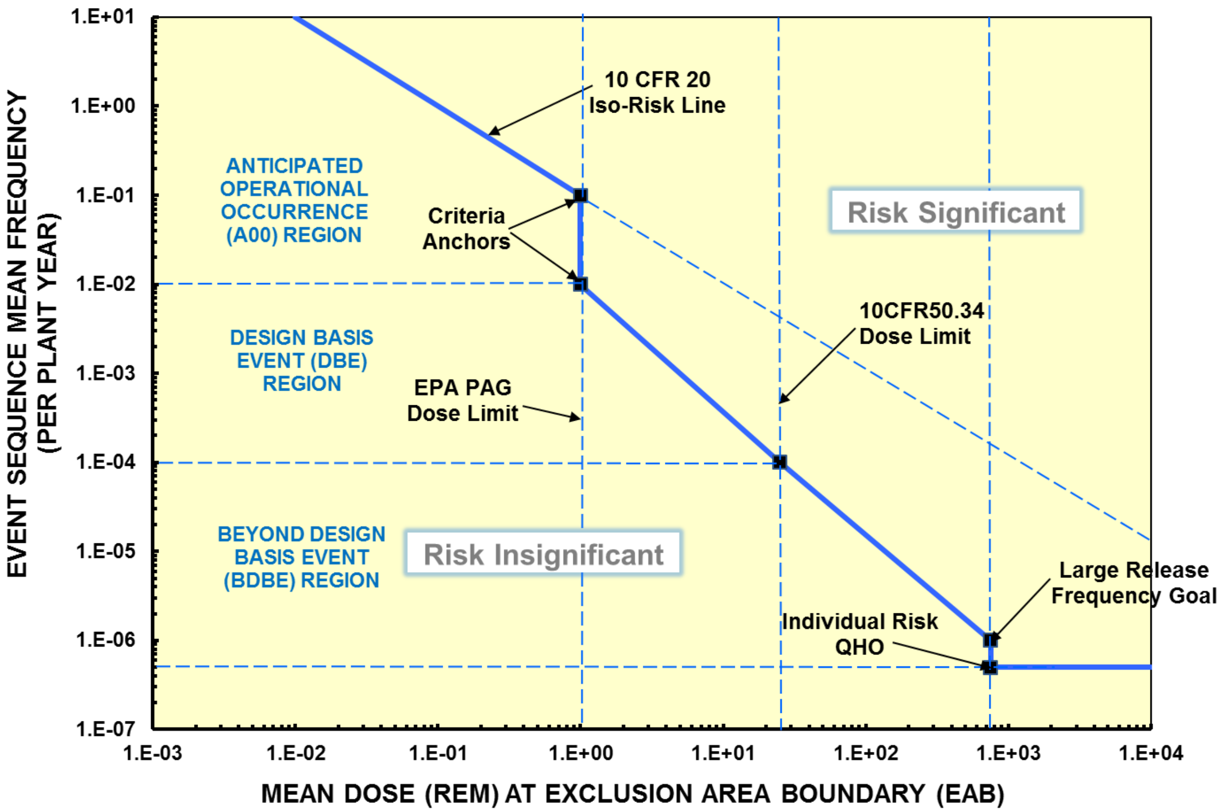

Figure ES-2 Frequency-Consequence Evaluation Criteria Proposed for LMP

Task 9 Proceed to Next Stage of Design Development
The decision to proceed to the next stage of design is reflected in this task. This implies not only completion of the design but also confirmation that defense-in-depth criteria evaluated in Task 7e have been satisfied.

Task 10 Finalize List of LBEs and Safety Related SSCs
Establishing the final list of LBEs and safety related SSCs signifies the completion of the LBE selection process and the selection of the safety related SSCs. The next step in implementing the TI-RIPB approach is to formulate performance requirements and regulatory design criteria for SSCs that are necessary to keep the LBE frequencies and doses within the TLRC frequency-dose criteria. Important information from Task 7b is used for this purpose.

MOTIVATION

The Commission's 1995 PRA Policy Statement states that a probabilistic approach to regulation enhances and extends the traditional deterministic approach by allowing consideration of a broader set of potential challenges to safety. This policy states:

> *"The Commission believes that an overall policy on the use of PRA methods in nuclear regulatory activities should be established so that the many potential applications of PRA can be implemented in a consistent and predictable manner that would promote regulatory stability and efficiency. In addition, the Commission believes that the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the state of the art in PRA methods and data and in a manner that complements the NRC's deterministic approach."*

The Policy Statement states further:

> *"A probabilistic approach to regulation enhances and extends this traditional, deterministic approach, by:*
> *(1) Allowing consideration of a broader set of potential challenges to safety,*
> *(2) Providing a logical means for prioritizing these challenges based on risk significance, and*
> *(3) Allowing consideration of a broader set of resources to defend against these challenges."*

The LBE selection and evaluation approach presented in this paper is guided by and is consistent with this policy

CONTENTS

FIGURES

TABLES

ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
ALARA	as low as reasonably achievable (U.S. term)
ALARP	as low as reasonably practicable (U.K. term)
AOO	anticipated operational occurrences
ATWS	anticipated transient without scram
BDBE*	beyond design basis event
CFR	Code of Federal Regulations
COL	Combined License
DBA	design basis accident
DBE*	design basis event
DOE	Department of Energy
EAB	exclusion area boundary
GEH	General Electric - Hitachi
GDC	general design criteria
HTGR	high temperature gas-cooled reactor
LBE*	licensing basis event
LMP	Licensing Modernization Project
LOCA	loss of coolant accident
LPZ	low population zone
LMP	Licensing Modernization Project
LWR	light water reactor
mHTGR*	modular high-temperature gas-cooled reactor
MHTGR	a specific prismatic mHTGR designed and developed by DOE with General Atomics as the lead vendor
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
NTTF	Near Term Task Force
PAG	Protective Action Guide
PBMR	pebble bed modular gas-cooled reactor design and developed by the South Africa vendor
PRA	probabilistic risk assessment
PRISM	Power Reactor Innovative Small Module Liquid-Metal Reactor
PSID	Preliminary Safety Information Document
QHO	quantitative health objectives

RAI	Request for Additional Information
RIPB	risk-informed and performance-based
SAP	Safety Assessment Principle
SAR	Safety Analysis Report
SBO	station blackout
SRP	Standard Review Plan
SSC	structures, systems, and components
TEDE	total effective dose equivalent
TI-RIPB*	technology inclusive, risk-informed, and performance-based
TLRC*	top-level regulatory criteria
UK	United Kingdom
YM	Yucca Mountain

*These terms have special meanings defined in this document

1. INTRODUCTION

1.1 Purpose

Many of the current regulatory requirements for US nuclear power plants are based on light water reactor (LWR) technology used for generation of electricity, necessitating changes to the LWR framework[2] to facilitate efficient, effective, and predictable licensing expectations for a spectrum of novel, advanced, non-LWRs. The Licensing Modernization Project (LMP), led by Southern Company and cost-shared by the U.S. Department of Energy (DOE) and other industry participants, is proposing changes to specific elements of the current licensing framework and a process for implementation of the proposals. These proposals are described in a series of papers (including this paper), which will collectively lead to modernization and adaptation of the current licensing framework to support licensing of advanced non-LWRs. These proposals are intended to retain a high degree of nuclear safety, establish stable performance-based acceptance criteria, and enable near-term implementation of non-LWR design development, in support of national and industrial strategic objectives. The LMP objective is to support NRC efforts to develop regulatory guidance for licensing advanced non-LWR plants.

These proposals are technology-inclusive, risk-informed, and performance-based (TI-RIPB). The modernized framework is technology-inclusive to accommodate the variety of technologies expected to be developed (implementation obviously will be technology-specific). It is risk-informed because it employs an appropriate blend of deterministic and probabilistic inputs to each decision. It is performance-based because it uses quantitative risk metrics to evaluate the risk significance of events and leads to formulation of performance requirements on the capability and reliability of structures, systems, and components to prevent and mitigate accidents. By utilizing a risk-informed, performance-based approach for the Licensing Basis Event selection process the design and licensing efforts are more closely aligned with the safety objectives. The goal is efficient and effective development, licensing, and deployment of non-LWRs on aggressive timelines with even greater margins of safety than prior generations of technology. These goals fully support and reflect DOE and US Nuclear Regulatory Commission (NRC) visions for licensing and deploying advanced non-LWR plants.

The new framework consists of elements including: establishment of TI-RIPB licensing-basis event selection; classification of structures, systems, and components; and establishment of predictable means to determine and preserve adequate defense in depth. These process steps are facilitated and informed by papers describing approaches and methods for: risk-informed decision making; the conduct and application of probabilistic risk assessments as part of the early and continuing lifecycle of new designs; and establishment of performance-based licensing criteria in lieu of LWR-centric prescriptive requirements. These elements are supported by reviews of past regulatory precedents and policies to make maximum use of existing approaches and NRC decisions, as well as assessments of current state of the art analytical tools. Gap analyses are used to identify where new or revised requirements are needed for a TI-RIPB framework and propose changes in language or approach to allow the framework changes to be used effectively.

The relationship between the main topics described above is represented in Figure 1-1. A simple diagram cannot capture these relationships comprehensively because the development process for a licensing framework is iterative, not serial; there are feedback loops that are difficult to represent in a simple figure,

[2] "Framework" as used in the LMP products, refers to the interrelated elements that form the basis for the NRC's oversight of the use of radioactive materials, including the Atomic Energy Act and enabling legislation; licenses, orders, and regulations in Title 10 of the Code of Federal Regulations; regulatory guides, review plans, and other documents that clarify and guide the application of NRC requirements and amplify agency regulations; and licensing and inspection procedures and enforcement guidance. The focus of the LMP effort is primarily on new or amended regulatory guidance and implementation proposals (i.e., near-term changes in actual regulation are not anticipated as part of LMP initiatives).

and some outputs are not shown. Nonetheless, this figure is intended to provide a generalized context for the major activities and how they fit into the overall framework.

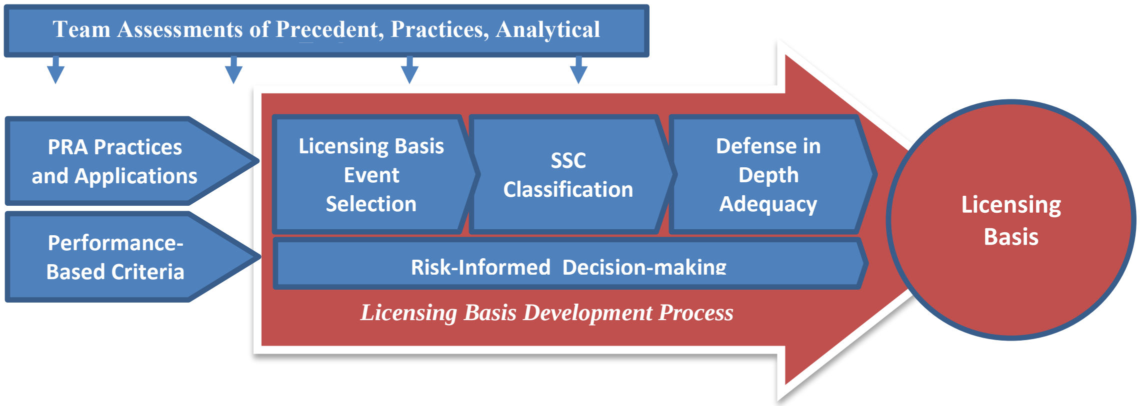

Figure 1-1 Elements of TI-RIPB Licensing Modernization Framework

This report, Modernization of Technical Requirements for Licensing of Non-Light Water Reactors, Licensing Basis Event Selection, represents a key element in development of a framework for the efficient licensing of advanced non-light water reactors (non-LWRs). It is the result of a project led by Southern Company and cost-shared by the United States Department of Energy (US DOE). This Licensing Modernization Project (LMP) will result in detailed proposals for establishing licensing technical requirements to facilitate efficient design and licensing of advanced non-LWRs. Such a framework acknowledges enhancements in the level of safety achievable with advanced designs. It also reflects current knowledge regarding safety and design innovation, creating an opportunity for reduced regulatory complexity without diminishing levels of safety. The project builds on best practices as well as previous activities through DOE and industry-sponsored advanced reactor licensing initiatives.

This white paper reviews the relevant regulatory precedents for guidance in identifying the spectrum of licensing basis events (LBEs) to be considered, describes the methodology for selecting and classifying LBEs, and sets forth issues for discussion in order to facilitate an effective submittal leading to license applications for advanced non-LWRs. This paper builds on the development and review of an LBE white paper for DOE's Next Generation Nuclear Plant (NGNP) and is intended for use with a spectrum of advanced non-LWRs including modular HTGRs, molten salt reactors, and liquid metal cooled fast reactors.

1.2 Objective of this Paper

The objective of this paper is to provide a technology-inclusive, risk-informed, and performance-based (TI-RIPB) approach for the selection of LBEs to support the preparation of license applications for advanced non-LWR plants. Included in this work is a systematic and reproducible process to achieve the following objective identified in the Standard Review Plan for Transient and Accident Analysis [3]:

> *"If the risk of an event is defined as the product of the event's frequency of occurrence and its consequences, then the design of the plant should be such that all the AOOs and postulated accidents produce about the same level of risk (i.e., the risk is approximately constant across the spectrum of AOOs and postulated accidents). This is reflected in the general design criteria (GDC), which generally prohibit relatively frequent events (AOOs) from resulting in serious*

consequences, but allow the relatively rare events (postulated accidents) to produce more severe consequences."

1.3 Scope

The approach described in this paper applies to a spectrum of advanced non-LWR designs including modular HTGRs (mHTGRs), molten salt reactors, liquid metal cooled fast reactors, and other known concepts and is intended to be reactor technology inclusive. This white paper discusses selection and classification of licensing basis events (LBEs) using criteria that focus on acceptable risks and consequences to the public. LBEs include all the events considered in the design and licensing of the plant and include the prescriptive design basis accidents (DBAs). Risks and consequences to the worker are also important, but will be discussed at a later date, as will security-related events.

Section 2 of this white paper provides an overview of the regulations and guidance considered during development of the proposed LBE selection and classification approach. This TI-RIPB approach is described in Section 3 and builds upon an approach that was developed for the U.S. Department of Energy's MHTGR[x] and Next Generation Nuclear Plant Projects [1] by incorporating lessons learned from NRC and ACRS reviews of that approach and by considering its application in a reactor technology inclusive manner. It also considers events and developments in the intervening period following the NGNP work, such as new insights from the Fukushima Accident, and additional NRC regulatory framework updates and studies. Section 3 includes a discussion of how both probabilistic and deterministic inputs are considered for informing the design and the events to be considered in licensing. Section 4 summarizes the top priority licensing topics to be discussed with the NRC staff and examines how the proposed approach for selecting LBEs meets the existing regulatory foundation in Section 2 and the guidance and precedents in this area.

1.4 Summary of Outcome Objectives

The LMP objective is to assist the NRC to develop regulatory guidance for licensing advanced non-LWR plants. This paper presents a modern, technology-inclusive, risk-informed, and performance-based (TI-RIPB) approach to identifying LBEs, which cover a spectrum of events considered in the design and licensing of a nuclear power plant. A key licensing outcome of this approach is the structured selection of design basis accidents (DBAs) that are traditionally analyzed in Chapter 15 of the license application.

In this paper, the LMP is seeking:

(1) NRC's approval of the proposed LBE selection approach for incorporation into appropriate regulatory guidance;
(2) Identification of any issues that have the potential to significantly impact the selection and evaluation of LBEs, including anticipated operational occurrences (AOOs), design basis events (DBEs), beyond design basis events (BDBEs) and design basis accidents (DBAs)

The proposed LBE selection approach covers license applications for a single reactor and multi-reactor module plants.[3]

The LMP is seeking NRC agreement on the following statements:

---

[3] *Plant*, as the term is used in this document means a nuclear plant that may or may not employ a *modular design*.

Modular design means a nuclear power plant that consists of two or more essentially identical nuclear reactors (modules) and each module is a separate nuclear reactor capable of being operated independent of the state of completion or operating condition of any other module co-located on the same site, even though the nuclear power plant may have some shared or common systems [2].

- The structured, TI-RIPB process described in this document is an acceptable approach for defining the LBEs for advanced non-LWRs such as modular HTGRs, molten salt reactors, and liquid metal cooled reactors. A means of documenting NRC review and approval of this approach is an essential outcome objective.

- The LMP approach to defining LBEs is broadly acceptable. As the term is used in this document, LBEs are defined broadly to include all the events used to support the safety aspects of the design and to meet licensing requirements. They cover a comprehensive spectrum of events from normal operation to rare off-normal events. There are four categories of LBEs:
  - Anticipated Operational Occurrences (AOOs), which encompass normal operation and planned and anticipated events whose frequencies exceed 10^{-2}/plant-year where a plant may be comprised of one or more reactor modules. The radiological doses from AOOs are required to meet normal operation public dose requirements. AOOs are utilized to set operating evaluation criteria for normal operation modes and states.
  - Design Basis Events (DBEs) encompass unplanned off-normal events not expected in the plant's lifetime whose frequencies are in the range of 10^{-4} to 10^{-2}/plant-year, but which might occur in the lifetimes of a fleet of plants. The radiological doses from DBEs are required to meet accident public dose requirements. DBEs are the basis for the design, construction, and operation of the structures, systems, and components (SSCs) during accidents.
  - Beyond Design Basis Events (BDBEs), which are rare off-normal events whose frequencies range from $5x10^{-7}$/plant-year to 10^{-4}/plant-year. BDBEs are evaluated to ensure that they do not pose an unacceptable risk to the public and to provide input to the selection of DBAs.
  - Design Basis Accidents (DBAs). The DBAs for Chapter 15, "Accident Analyses," of the license application are prescriptively derived from the DBEs by assuming that only SSCs classified as safety-related are available to mitigate the consequences. The public consequences of DBAs are based on mechanistic source terms and are conservatively calculated. The upper 95% conservative estimate of the dose of each DBA must meet the 10 CFR §50.34 consequence limit at the Exclusion Area Boundary (EAB). The DBAs are not selected on the basis of frequency, but rather by a set of prescriptive rules similar to those employed in defining DBAs for existing LWRs. As shown with examples in this report for two types of advanced non-LWRs, they often correspond to event sequences modeled in the PRA with extremely low frequencies.

  The LMP technology- inclusive, risk-informed, and performance- based (TI-RIPB) approach to selecting LBEs is designed to ensure that an appropriate set of limiting events for each reactor technology are reflected in the selection of DBAs and that the full set of LBEs define the risk significant events for each design and technology. This is essential to ensure that risk insights are appropriately reflected in the design and licensing decisions.

  The LBEs in each category are evaluated individually to support the tasks of assessing the performance of SSCs with respect to safety functions in response to initiating events to meet applicable regulatory limits and collectively to demonstrate that the integrated risk of a multi-reactor module plant design meets the NRC Safety Goals. There will be different LBEs for events affecting single and multiple reactor modules. An important outcome of the selection and evaluation of LBEs is to identify design features of the plant that are necessary and sufficient to ensure that risk goals in the NRC Safety Goal Policy are achieved and licensing requirements are met. The use of these insights in the derivation of performance requirements and principal design criteria for SSCs, including the radionuclide barriers, is a topic of a future LMP white paper on SSC safety classification.
- Implementation of the proposed TI-RIPB approach to selecting LBEs requires the development of deterministic and probabilistic inputs to the LBE selections that have sufficient technical adequacy to

support such decisions. The approach to performing the required PRA inputs and for achieving the necessary technical adequacy of the PRA is the topic of a companion LMP deliverable to be provided for review. The PRA is introduced at an early stage of the design to support design decisions and the level of detail and scope of the PRA is consistent with the level of detail of the design and site characterization.

- In order to address the selection of LBEs for a plant with two or more reactor modules or radionuclide sources[4], the frequencies of LBEs are expressed in units of events per plant-year where a plant is defined as a specific collection of reactor modules within the scope of the license application[5]. Thus, each LBE may involve a plant response or release from one or multiple reactors or radionuclide sources The evaluation criteria on the frequency ranges for the LBE categories are as follows:
  - AOOs – event sequences with mean frequencies greater than 10^{-2} per plant-year
  - DBEs – event sequences with mean frequencies less than 10^{-2} per plant-year and greater than 10^{-4} per plant-year
  - BDBEs – event sequences with mean frequencies less than 10^{-4} per plant-year and greater than 5×10^{-7} per plant-year.
  - DBAs –are deterministically defined and are not selected on the basis of frequency. However, the plant response to each DBA corresponds to either a DBE, BDBE or lower frequency accident sequence.

- Acceptable offsite dose evaluation criteria on the event sequence consequences for the LBE categories are defined by a frequency-consequence evaluation criteria derived from Top Level Regulatory Criteria (TLRC). The TLRC frequency-consequence criteria are used to evaluate the risk significance of each LBE. Key elements of the TLRC used to develop the frequency-consequence criteria include:
  - AOOs – 10 CFR Part 20: 100 mrem total effective dose equivalent (TEDE) mechanistically modeled and realistically calculated at the exclusion area boundary (EAB). For the advanced non-LWR facilities, the EAB is expected to be the same area as the controlled area boundary.
  - DBEs – 10 CFR §50.34: 25 rem TEDE mechanistically modeled and realistically calculated at the EAB.
  - BDBEs – NRC Safety Goals for large release frequency and quantitative health objectives (QHOs) for the risk of individual fatality are mechanistically and realistically calculated out to 1 mile (1.6 km) from the site boundary for early health effects and 10 miles (16 km) from the site boundary for latent health effects.

- In addition to evaluating the risk significance of individual LBEs, the LMP approach to evaluating LBE includes several criteria to ensure that the integrated risk of the advanced non-LWR plant, which may be comprised of two or more reactor modules, is acceptably small and consistent with the NRC Advanced Reactor and Safety Goal policies. These criteria include:
  - The total frequency of exceeding of a site boundary dose of 100 mrem shall not exceed 1/plant-year to ensure that the annual exposure limits in 10 CFR 20 are not exceeded.
  - The total frequency of a site boundary dose exceeding 750 rem shall not exceed 10^{-6}/plant-year. Meeting this criterion would conservatively satisfy the NRC Safety Goal Policy Statement [48] on limiting the frequency of a large release.

---

[4] Non-reactor sources include spent fuel storage, fuel processing, and rad-waste processing and storage systems.

[5] Each reactor module may be separately licensed, but when the second and subsequent modules are licensed the multi-module LBEs will be defined, and the plant capabilities to ensure that multi-module accident risks are not significant will be incorporated into the licensing basis.

    - The average individual risk of early fatality within the area 1 mile of the EAB shall not exceed $5x10^{-7}$/plant-year to ensure that the NRC Safety Goal QHO for early fatality risk is met.
    - The average individual risk of latent cancer fatalities within the area10 miles of the EAB shall not exceed $2x10^{-6}$/plant-year to ensure that the NRC safety goal QHO for latent cancer fatality risk is met.

- The frequency below which events are not selected as BDBEs is 5×10^{-7} per plant-year. Satisfaction of the NRC safety goal QHOs is assured when this frequency is not exceeded. The PRA examines events to 10^{-8} per plant-year to assure that there are no "cliff edge effects" just below this de minimus frequency.

- The kinds of events, failures, and natural phenomena that are evaluated include:
    - single, multiple, dependent, and common cause failures to the extent that these contribute to LBEs and their frequencies
    - events affecting one or more than one reactor module or radionuclide source within the scope of the license application
    - internal events (including transients and accidents) and internal and external plant hazards that occur in all operating and shutdown modes and potentially challenge the capability to satisfactorily retain sources of radioactive material.

- Uncertainty distributions including upper and lower 95% confidence values are evaluated for the frequency and the consequence for each AOO, DBE, and BDBE.
    - the mean frequency is used to determine whether the event sequence family is an AOO, DBE, or BDBE. If the upper or lower bound on the LBE frequency straddles two or more regions, the LBE is compared against the frequency and consequence criteria for each region.
    - sources of uncertainty that are identified by the PRA and not fully resolved via quantification are addressed as part of a risk-informed evaluation of defense-in-depth as addressed in a companion LMP deliverable on defense-in-depth.
    - The mean consequences are explicitly compared to the consequence criteria in all applicable LBE regions.
    - The upper bound consequences for each DBA, defined as the 95%tile of the uncertainty distribution, shall meet the 10 CFR §50.34 dose limit at the EAB. Sources of uncertainty in both frequencies and consequences of LBEs are identified and addressed in the LMP approach to defense-in-depth.

1.5 Relationship to Other LMP Pre-Licensing Topics/Papers

This white paper is one of several LMP products covering key regulatory issues that are being prepared and submitted for NRC review for the purpose of establishing regulatory guidance for advanced non-LWR developers. Some of these issues have a bearing on the development of the methodology for selecting LBEs or will rely on the process outlined in this paper. The topics that are planned to be addressed within the scope of the LMP include:

- LMP Approach to PRA Development for Licensing Basis Event Selection
- LMP Approach to PRA for RI-PB risk management applications
- LMP SSC Safety Classification and Performance Requirements Approach
- LMP Defense-in-Depth Adequacy

2. REGULATORY FOUNDATION AND PRECEDENTS

There is a substantial set of prior activities, policies, practices and precedents stretching more than 30 years back in time that inform RIPB processes and uses. NRC and international regulations, policies, guidance, and other precedents that are relevant to the definition of LBEs and their treatment are discussed in this section. NRC and ACRS feedback on previous efforts to define LBEs for Advanced Non-LWRs are also reviewed for LBE definition guidance. This regulatory background is examined to investigate two aspects of the proposed TI-RIPB approach for the LMP project. The first is the process of defining and selecting the LBEs and the second is the development of the Top-Level Regulatory Criteria (TLRC) that are used to establish evaluation boundaries on the frequencies and radiological consequences for classifying and evaluating the LBEs. The scope of this review includes U.S. regulatory requirements as specified in the regulations, and supporting policies, Commission directives, regulatory guidance, and Standard Review Plan as well as international safety standards. Insights from NRC pre-licensing reviews of advanced non-LWRs are also included. This section of the white paper builds on the regulatory review in the NGNP White Paper on LBE selection [1] by incorporating more recent developments and precedents and by considering the need to have a reactor technology inclusive approach for selecting LBEs rather than one focused on HTGR-specific technology only. Observations and conclusions reached from this review that are used in the definition of the LBE approach are summarized at the end of this section.

2.1 Regulatory Foundation and Precedent Review Summary

This section reviews NRC requirements and other relevant precedents for insights on how to select LBEs for a new reactor design. This review reflects on the qualitative approach to risk used in the past, relying on judgment and prescription derived from years of LWR design, analysis and operations. The purpose is not to criticize, but rather to identify desirable attributes of a TI-RIPB approach to the selection of LBEs.

NRC regulatory requirements for the design of currently licensed and new reactors refer to several different kinds of events included within the licensing basis including anticipated operational occurrences (AOOs), design basis events (DBEs), postulated accidents, design basis accidents (DBA), and beyond design basis events (BDBE). The definitions of these events are similar to LBE types introduced in Section 1.4 however there are significant differences in licensing event terminology as shown in Table 2-1.

For normal operations, including AOOs, the NRC regulations are, for the most part, generic and appear to generally apply to an advanced non-LWR plant. The applicant is required to classify the events considered within the design basis as either AOO or accident (DBA) based on a qualitative and presumably subjective assessment of the expected frequency of occurrence because there are no quantitative frequency criteria included. In many cases it is unclear whether the qualitative characterization of frequency refers to that for an initiating event or for an entire accident sequence. While the applicant's classification is subjected to NRC staff review there is no quantification of the event frequencies nor a prescribed method for ensuring that design specific events are adequately considered. A concern for advanced non-LWRs is that events that are uniquely appropriate for a given reactor technology are likely not represented on the supplied lists of generic LWR events, so it is necessary to have a method that is systematic and reproducible to derive the appropriate list of LBEs. For non-LWR plants whose designs depart in major ways from those of existing and even advanced non-LWRs, a more systematic and quantitative means of identifying the unique events and correctly classifying their frequencies would be necessary to ensure a safe design and contribute to a more predicable path to a license.

Table 2-1 Definitions of Licensing Basis Events

Event Type	NRC Definition	LMP Definition
Anticipated Operational Occurrences (AOOs)	*"Conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit*[6] *and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power."* [SRP 15.0 and 10 CFR 50 Appendix A]	Conditions of plant operation, events, and event sequences that are expected to occur one or more times during the life of the nuclear power plant which may include one or more reactor modules. Events and event sequences with frequencies of $1x10^{-2}$ per plant year and greater are classified as AOOs. AOOs take into account the expected response of all SSCs within the plant regardless of safety classification.
Design Basis Events (DBEs)	*"Conditions of normal operation, including AOOs, design-basis accidents, external events, and natural phenomena, for which the plant must be designed to ensure functions of safety-related electric equipment that ensures the integrity of the reactor coolant pressure boundary; the capability to shut down the reactor and maintain it in a safe shutdown condition; or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures."* [SRP 15.0]	Events and event sequences that are expected to occur one or more times in the life of an entire fleet of nuclear power plants, but less likely than an AOO. Events and event sequences with frequencies of $1x10^{-4}$ per plant year to $1x10^{-2}$ per plant year are classified as DBEs. DBEs take into account the expected response of all SSCs within the plant regardless of safety classification. The objective and scope of the DBEs to form the design basis of the plant is the same as in the NRC definition. However, DBEs do not include normal operation and AOOs as defined in the NRC references.
Beyond Design Basis Event (BDBE)	"*This term is used as a technical way to discuss accident sequences that are possible but were not fully considered in the design process because they were judged to be too unlikely. (In that sense, they are considered beyond the scope of design-basis accidents that a nuclear facility must be designed and built to withstand.) As the regulatory process strives to be as thorough as possible, "beyond design-basis" accident sequences are analyzed to fully understand the capability of a design."* [NRC Glossary]	Events and event sequences that are not expected to occur in the life of an entire fleet of nuclear power plants. Events and event sequences with frequencies of $5x10^{-7}$ per plant year to $1x10^{-4}$ per plant year are classified as BDBEs. BDBEs take into account the expected response of all SSCs within the plant regardless of safety classification. The objective of BDBEs to assure the capability of the plant is the same as in the NRC definition.

[6] SRP 15.0 further breaks down AOOs into events with "moderate" frequency (events expected to occur several times during the plant life) and "infrequent" (events that may occur during the plant life)

Event Type	NRC Definition	LMP Definition
Design Basis Accidents (DBA)	*"Postulated accidents that are used to set design criteria and limits for the design and sizing of safety-related systems and components."* [SRP 15.0] "*A postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety*." [NRC Glossary and NUREG 2122]	Postulated accidents that are used to set design criteria and performance objectives for the design and sizing of SSCs that are classified as safety-related. DBAs are derived from DBEs and high consequence BDBEs based on the capabilities and reliabilities of safety related SSCs needed to mitigate and prevent accidents, respectively. DBAs are derived from the DBEs by prescriptively assuming that only SSCs classified as safety related are available to mitigate postulated accident consequences to within the 50.34 dose limits.
Licensing Basis Events (LBEs)	Term not used formally in NRC documents	The entire collection of events considered in the design and licensing basis of the plant, which may include one or more reactor modules. LBEs include normal operation, AOOs, DBEs, BDBEs, and DBAs

Moreover, establishing an appropriate set of reactor technology specific LBEs cannot wait until the submittal of a license application. This selection is essential to the development of any design and must be established very early in the design process.[7]

All the example events given in the definition of AOO in the regulations and in the supporting regulatory guides and Standard Review Plan [3] are applicable to LWRs. While some of these may apply, many may not be applicable to a particular non-LWR design.

In the selection of LBEs it is expected that the selection will consider a comprehensive and exhaustive set of events from which to identify the "limiting" events. However, specific criteria for how to determine which events are limiting are not provided in existing regulatory guidance. In addition, it is not clear from the regulatory guidance which events are considered to be limited by the selected events. This points to a need for a systematic and reproducible process to identify the DBAs for the deterministic safety analysis.

With few exceptions, such as provisions for protection against natural phenomena and inclusion of some generic events in the lists of example events such as loss of offsite power and station blackout, the regulations that have evolved for unplanned transients and accidents are light water reactor (LWR)-specific. The GDC define the types of design considerations that apply to the design of SSCs that prevent or mitigate a specified set of postulated accidents. For example, GDC typically indicate that safety systems must be able to perform their design basis functions given a single active failure and a concurrent loss of offsite power.

NRC's regulations do not have performance-based criteria to limit the consequences of BDBEs nor quantitative criteria for classifying events as BDBEs based on frequency other than noting they were considered too infrequent to be included in the design basis. In apparent response to events that have occurred but had not been anticipated in the original design and licensing bases, regulations have been added to provide protection against selected BDBEs. Examples of these include: anticipated transients without scram (ATWS) addressed in 10 CFR §50.62 [5], station blackout addressed in 10 CFR §50.63.

The regulations associated with licensing events and their supporting regulatory documents do not distinguish well between events and event sequences for the purpose of characterizing the frequency of occurrence and classifying as either an AOO, DBA, or BDBE. The term "sequence of events" is referred to here in the context of analyzing how the plant responds to initiating events. The point here is a given event may be characterized at a certain frequency level and severity of plant impact, but when compounded by additional failures both the frequency and the level of impact are different. Hence, there may be different LBEs having different levels of frequency and severity stemming from the same initiating event. In reviewing the regulatory documents, it is extremely difficult to sort out in most cases whether the term "events" refer to initiating events only or to some sequence of events. A goal of the LMP is to consider initiating events and the associated event sequences as distinct challenges to the safety functions in order to provide sufficient completeness in the identification of LBEs.

In many cases the events classified as AOOs or DBAs as discussed in the regulations and supporting SRP are referred to as "initiating events". By applying the single failure criterion, the safety analysis for the DBAs includes the requirement that the "worst" active single failure be assumed in demonstrating that

---

7 One additional definition is required to understand the importance of the terms. The *Design bases* means that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state of the art" practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals. [10CFR50.2 Definitions]

safety criteria are met, however the probability of the single failure does not have any bearing on the classification of the event. In addition, non-safety related SSCs including offsite power supplies are assumed not to be available in the deterministic safety analysis of DBAs, which also would be considered if the frequency of the DBA were to be assessed.

With few exceptions, there does not appear to be a consideration of the probability of a common cause failure that could occur in combination with an initiating event to produce a DBA even though the service experience indicates that there have been many occurrences of such events. The application of the single failure criterion for DBAs seems to assume that common cause failures will be prevented by meeting the design requirements. In the limited cases of selected BDBE requirements, such as those for anticipated transients without scram ATWS and station blackout (SBO), event sequences that could be caused in part by a common cause failure and involve multiple failures of redundant components are identified as being comprised of a sequence initiated by an AOO (transient for ATWS and loss of offsite power for SBO). However, a systematic way to consider both events and event sequences that could be comprised of combinations of single failures and common cause failures is not included in the enumeration of prescriptive events nor in the characterization of their frequencies or level of severity of the challenge.

An important insight from this review is that, based on what can be gleaned from the regulations and supporting documents, the historical approach to selecting an appropriate set of LBEs for a given design is ad hoc. The challenge facing designers and licensees for advanced non-LWRs is to find a process for selecting LBEs that is systematic, reproducible, and capable of identifying the appropriate limiting events for a given design.

2.2 Summary of Documents Reviewed

A summary of the documents reviewed for regulatory guidance and insights from relevant precedents is provided in Table 2-2. The regulatory documents include the U.S. Code of Federal Regulations, NRC policies and policy statements, NRC Staff Requirements Memoranda, regulatory guides, the Standard Review Plan (NUREG-0800), and relevant Advisory Committee on Reactor Safeguards letters. The relevant regulatory precedents include the initiatives to develop RIPB licensing approaches for the MHTGR, PRISM, PBMR, and the Department of Energy Next Generation Nuclear Plant (NGNP) and the NRC staff and ACRS reviews and feedback on those initiatives.

Additional insights were developed by reviewing the use of PRA to perform the Pre-Closure Safety Analysis which was required for licensing the Yucca Mountain repository as well as the frequency-consequence criteria that were incorporated into the regulations for that facility. International perspectives were incorporated into the review based on relevant documents from the IAEA and the regulatory authority in the United Kingdom. A full discussion of the LBE selection and evaluation insights derived from these documents is found in Appendix A. The conclusions from this review are presented in the next section.

2.3 Precedent Review Summary

The following observations and conclusions are made in this review of the regulatory foundation for selection of LBEs for advanced non-LWRs. These observations and conclusions shape the development of an approach for LBE selection that is provided in Section 3 of this white paper.

Table 2-2 Documents Reviewed for Regulatory Bases and Precedents

Category	Reference	Applicable content[1]
NRC Regulations	10 CFR Part 50 Appendix A	AOO definition
	10 CFR Part 50.34	Dose limits for postulated accidents
	10 CFR Part 50.44	Requirements for combustible gas control
	10 CFR Part 50.62	Requirements for ATWS
	10 CFR Part 50.63	Requirements for SBO
	10 CFR Part 50.150	Requirements for aircraft impact assessment
	10 CFR Part 52.1	Definitions for reactor unit, modular design
	10 CFR Part 63	Frequency and dose performance requirements for Yucca Mountain Pre-closure Safety Analysis
	10 CFR Part 63.111	Performance objectives for geologic repository
	10 CFR Part 63.112	Performance objectives for pre-closure operations
	10 CFR Part 20	Annual dose limits for normal operation and AOOs
	10 CFR Part 50 Appendix I	Design objectives for keeping releases ALARA
	10 CFR Part 52.79	Principal design criteria for SSCs to limit doses
	10 CFR Part 100	Dose limits for defining EAB and LPZ
	40 CFR Part 190	Environmental radiation protection standards
NRC Policies	73 FR 60612	Policy on regulation of advanced reactors
	60 FR 42622	Policy on use of PRA
	51 FR 28044	Safety goal policy
	50 FR 32138	Severe accident policy
NRC Policy Statements	SRM/SECY 90-16	Evolutionary LWR certification issues
	SECY 2002-0076	Semi-annual update on future licensing
	SECY 2003-0047	Policy issues related to non-LWR licensing
	SRM 2003-0047	Staff requirements memorandum for SECY 2003-0047
	SECY 2005-0006	Regulatory structure and policy issues for new plant licensing
	SECY 2010-0034	Policy, licensing and technical issues for SMRs
	SECY 2011-0079	License structure for multiple module SMRs
	SECY 2011-0152	Emergency planning for SMRs
	SECY 2013-0029	History of large release frequency metric
	SECY 2016-0012	Accident source terms for SMRs and non-LWRs
NRC Guidance	NUREG-0800, Chapter 15	Transient and Accident Analysis
	NUREG-0800, Chapter 19	PRA and severe accident evaluation
	Reg. Guide 1.174	Use of PRA in risk-informed decisions approach
	Reg. Guide 1.200	Technical adequacy of PRA
	NUREG/BR-0303	Performance-based regulation guidance
	NUREG-1860	RIPB regulatory structure feasibility study
	NUREG-2150	Proposed risk management regulatory framework
	NRC NTTF Report	Review of Fukushima Daiichi accident
ACRS	ACRS letter April 22,2004	ACRS views on risk metrics for non-LWRs and interpretation of safety goal QHOs
NGNP	INL/EXT-09-17139	Defense-in-Depth White Paper
	INL/EXT-10-19521	Licensing Basis Event White Paper
	INL/EXT-11-21270	PRA White Paper
	INL/EXT-13-28205	NRC licensing status summary

Category	Reference	Applicable content[1]
	ACRS Letter May 15, 2013	ACRS views on NGNP proposed licensing approach
PBMR	Exelon Letter March 15, 2002	PBMR RIPB licensing approach
	NRC Letter Sept. 24, 2007	RAIs regarding PBMR white papers
	PBMR Letter March 21, 2008	Response to RAIs from Sept. 24, 2007
	NRC Letter March 26, 2002	NRC preliminary findings on licensing approach
MHTGR	DOE-HTGR-86-024	Preliminary safety information for MHTGR
	DOE-HTGR-86-011	PRA for MHTGR
	DOE-HTGR-86-034	Licensing basis events for MHTGR
	NUREG-1338	Draft Pre-application safety evaluation for MHTGR
PRISM	NUREG-1368	Pre-application safety evaluation for PRISM
	GEH 2017 report	Development and modernization of PRISM PRA
Yucca Mountain	DOE/RW-0573	Yucca Mountain Repository Safety Analysis Report
	NUREG-2108	NRC technical evaluation of YM SAR
	NUREG-1804	Yucca Mountain review plan
Industry Consensus Standards	ASME/ANS RA-Sb-2013	PRA standard for operating LWR plants
	ASME/ANS RA S-1.4-2013	Trial use PRA standard for advanced non-LWR plants
	ANS/ANSI-53.1-2011	Nuclear safety design process for modular helium cooled reactors
International Guidance	IAEA NSR-1	Nuclear safety design requirements
	UK SAPs	United Kingdom Safety Assessment Principles
	Farmer 1967 Paper	Proposal for a frequency-consequence risk criterion

[1] Acronyms used in table:

AOO Anticipated operational occurrence
ATWS Anticipated transient without scram
BDBE Beyond design basis event
ALARA As low as reasonably achievable
ACRS Advisory Committee on Reactor Safeguards
CFR Code of Federal Regulations
DOE Department of Energy
GEH GE Hitachi
HTGR High Temperature Gas-Cooled Reactor
MHTGR Modular HTGR
LWR Light water reactor
NTTF Near Term Task Force
PBMR Pebble bed modular reactor
PRA Probabilistic risk assessment
PRISM Power Reactor Innovative Small Module liquid metal reactor
RIPB Risk-informed and performance-based
SAPs Safety Assessment Principles
SAR Safety analysis report
SBO Station blackout
UK United Kingdom
YM Yucca Mountain

- Existing NRC Policy and Strategy statements fully support the greater use of RIPB practices. This vision is clearly articulated in NUREG-2150. There has been partial development of RIPB methods for the backfit, operation, oversight and modification of existing LWRs, however, little or no guidance for RIPB decision-making has been established for new, non-LWR advanced designs.
- The current U.S. regulations and regulatory guidance ("framework") for LWR-based designs do not include or provide a reproducible approach for selecting LBEs for advanced non-LWRs nor for ensuring that advanced non-LWRs of differing designs would be treated in a consistent manner for establishing their design and licensing bases.
- The only reactor technology inclusive set of regulatory documents that was identified in this regulatory review is that reflected in the U.K. Safety Assessment Principles (SAPs). The SAPs include numerical targets for evaluating LBE frequencies and consequences, which differentiate between those to be applied to each reactor unit and those that apply to the site as a whole. Different targets are expressed for regulatory evaluation boundaries and design objectives, thereby capturing the notion that risk are not to be used a strict pass-fail acceptance test.
- The approach that was developed for the MHTGR, and advanced for the Exelon, PBMR and NGNP project, as well as the approach used for PRISM for LBE selection, provide an appropriate baseline from which to develop the LBE selection process for advanced reactor design and licensing. An LBE selection approach proposed in NUREG-1860 was also reviewed for insights to help define desirable attributes of an effective LBE selection process. This regulatory foundation review provides guidance for refining and advancing these approaches.
- The RIPB approach advanced in the MHTGR, PBMR and NGNP projects has been reflected in a design standard for MHRs in ANS 53.1. This standard provides specific design criteria for implementing the approach that is consistent with the approach described in the NGNP white papers. These include criteria for evaluating the adequacy of defense-in-depth which contributes to the deterministic input to RIPB design decisions.
- There are a number of international precedents, including those from the U.S., IAEA, and the U.K. SAPs, and reflected in the NRC reviews of MHTGR, PRISM, and NGNP, that support the view that LBE selection is best accomplished through a risk-informed and performance-based process which includes both deterministic and probabilistic inputs and preserves the principle of defense-in-depth.
- A key challenge of any LBE selection process is to systematically define the initiating events that are appropriate for the reactor design, and the event sequences that realistically model the plant response to the initiating events. This is necessary in order to derive the appropriate and limiting Design Basis Accidents (DBAs) for that design. Simply removing inapplicable events from existing LWR events is not sufficient to define the events that are uniquely appropriate for a given design.
- The LBE definition and selection process must be clear in making the distinction between initiating events and event sequences. A given initiating event may result in different event sequences each having a different frequency of occurrence and level of severity in challenging the reactor safety defenses. Simply assuming the "worst active single failure" and concurrent loss of offsite power in combination with an initiating event does not necessarily yield the appropriate limiting accidents to define the licensing basis.

- As emphasized in NUREG-1860, PRA plays an important role in the identification and evaluation of uncertainties in the definition of event sequences and in the estimation of their frequencies and consequences. This information on sources of uncertainty and their influences on the risk assessment are important inputs to establishing adequate consideration of the principles of defense-in-depth in the selection and evaluation of LBEs and other RIPB decisions.
- In order to provide the technical basis for managing the risks of accidents that involve two or more reactors or radionuclide sources, by preventing and mitigating such accidents, it is necessary to consider such accidents in the definition of LBEs and to measure frequencies on a per (multi-reactor module) plant[8]-year basis, rather than reactor-year basis.
- The development of TLRC frequency-consequence criteria for the LMP project greatly benefits from the approach most recently advanced in the NGNP LBE white paper as well as similar frequency-consequence criteria originally proposed by Farmer. Useful guidance is also available from NUREG-1860 the U.K. SAPs for event consequences, frequencies and threshold for event evaluation.
- A key challenge in interpreting the current U.S. regulations for limiting radiological exposures for normal operation and LBEs is the lack of explicit numerical criteria for categorizing events by expected frequency of occurrence. However, the classification of LBEs into Anticipated Operational Occurrences (AOOs), Design Basis Events (DBEs), and Beyond Design Basis Events (BDBEs) based on expected frequency of occurrence is consistent with LBE classifications that were identified in this regulatory review including NGNP, PRISM, NUREG-1860, NUREG-2150, Yucca Mountain Pre-closure Safety Analysis, and the U.K. SAPs.
- There are a number of NRC criteria that explicitly constrain the risk and/or allowable consequences of radiological releases from nuclear power plants. These criteria include requirements to evaluate the adequacy of the proposed design of the plant against specific criteria. Some of the regulatory dose requirements are intended for evaluation of individual events, whereas others are expressed in terms of annual exposure limits, frequency of a given magnitude of release, and individual risks for the population in the vicinity of the plant site. The review of these criteria that was performed in the NGNP LBE White Paper [1] has been extended in this white paper and has yielded some new insights that are reflected in the proposed LBE selection approach as discussed in the next section.

The above key points have been used to guide the development of the LBE selection process as discussed more fully in Section 3.

---

[8] *Plant*, as the term is used in this document means a nuclear plant that may or may not employ a *modular design*.

Modular design means a nuclear power station that consists of two or more essentially identical nuclear reactors (modules) and each module is a separate nuclear reactor capable of being operated independent of the state of completion or operating condition of any other module co-located on the same site, even though the nuclear power station may have some shared or common systems [2].

3. PROPOSED ADVANCED non-LWR LBE SELECTION APPROACH

The regulatory bases reviewed in Section 2 of this white paper included two specific approaches for selecting licensing basis events (LBEs) for advanced non-LWRs including the approach originally employed for the MHTGR [40], subsequently refined in the Exelon PBMR [47] and NGNP projects [1], and subsequently used as a basis for ANS 53.1 , the design standard for modular HTGRs [66]. The other approach is that described in NUREG-1860 which has parallels to the approach proposed for HTGRs and offers additional guidance on the desirable features of an LBE selection approach. In addition to these resources, the regulatory review identified many other documents that provide expectations and useful guidance in selecting an LBE selection approach for the LMP project.

This section begins by listing the desirable attributes of an LBE selection process for advanced non-LWRs followed by reviews and observations from the supporting regulatory guidance. This review sets the stage for describing the proposed LBE selection process which is described in the balance of this section together with examples that have been prepared based on two advanced non-LWR designs including the MHTGR and the PRSIM.

3.1 LBE Selection Process Attributes

On the basis of the lessons learned from the regulatory bases in Section 2 and the objectives of the LMP, the desirable attributes of the LBE selection process for advanced non-LWRs should be:

Systematic and Reproducible: In principle, application of the process by different persons given the same inputs would yield a reasonably comparable set of LBEs. Any variations should only result from different states of knowledge that are fed into the process.

Sufficiently Complete: The LBE selection process should be capable of defining a sufficiently complete set of LBEs that is capable of defining the challenges to safety functions, radionuclide barriers, and protective strategies for emergency planning and accident management. In order to support the development of strategies to prevent and mitigate accidents involving multiple reactor modules and radionuclide sources, as occurred during the Fukushima Daiichi accident, and to enable the NRC review of design features responsible for implementing these strategies, the LBE selection process should address multi-module and multi-source accidents.

Available for Timely Input to Design Decisions: Importantly, the LBE selection process should recognize that design decisions that are impacted by LBE selection are made at an early stage of design and long before the licensing application is prepared. A key obstacle in limiting the progress in deploying advanced reactor technologies is the lack of predictability of licensing decisions. The LBE selection process should play an important role to support the optimization of the design with respect to safety.

Risk-informed and Performance-Based: The LBE selection process should be risk-informed and performance-based consistent with LMP objectives. Risk-informed, as contrasted with risk-based, means that the process will include an appropriate balance of deterministic and probabilistic elements, and will be consistent with the principles of defense-in-depth. Performance-based means that the process will include measurable and quantifiable performance metrics and will be consistent with NRC policies on use of performance-based alternatives. The interfaces with other risk-informed and performance-based decisions such as SSC safety classification, definition of SSC requirements for capability and reliability, and implementation of defense-in-depth strategies should be clearly defined.

Reactor Technology Inclusive: When applying the process to different advanced non-LWRs having fundamentally different safety design approaches will yield an appropriate set of LBEs that are consistent and fairly defined across the different reactor technologies. Appropriate means that the LBEs are capable of identifying the unique safety issues for each technology. Specifically, the approach needs to support a consistent definition of LBEs for modular HTGRs, molten salt reactors, and liquid metal cooled reactors using both thermal and fast neutron spectra and employing different safety design approaches. The LBE selection process should yield a uniform level of safety consistent with NRC safety goal and advanced reactor policies.

Consistent with Applicable Regulatory Requirements: The LBE selection process must account for the current regulatory requirements with due regard to their applicability to advanced non-LWR technologies and associated safety design approaches.

These attributes are consistent with the objectives of the risk-informed approach to selecting LBEs documented in NUREG-1860.

3.2 Review of Previous LBE Selection Approaches

The regulatory and precedent review summarized in Section 2 of this white paper identified two approaches that have been proposed to select LBEs for advanced non-LWRs. One of these, which will be referred to as the NGNP approach, was originally developed and applied to the MHTGR [40] and was subsequently refined in the pre-licensing interactions with the Exelon PBMR project [47], and further refined in the NGNP project [1]. The second is that described in NUREG-1860 [21]. In addition to these, the Yucca Mountain Pre-Closure Safety Analysis may be considered to be a third method for deriving LBEs using a risk-informed and performance based process. In this case the method is for a non-reactor facility governed by a different set of regulatory requirements. Because the NGNP approach has actually been applied to a conceptual non-LWR design that was supported by a PRA and subjected to review by NRC and supporting National Laboratories, the LBE selection process adopted for use in the LMP project is developed starting with a review of that approach. The proposed LBE selection process is then developed from this review utilizing insights from a review the NUREG-1860 approach and the regulatory precedent reviews for proposing selected refinements. The goal of this review is to define an LBE selection process that has the attributes presented in the previous section.

The NGNP LBE pproach utilizes Top Level Regulatory Criteria to define frequency vs. consequence criteria for evaluating the risks associated with LBEs as shown in Figure 3-1. The key elements of the criteria are summarized as follows:

- LBEs are initially defined by accident families from the results of the PRA where each family has a similar initiating event, challenge to safety functions, plant response, and mechanistic source term for those families involving a release. LBEs are classified as AOOs with mean frequencies of 10^{-2}/plant-year or greater, DBEs with mean frequencies between 10^{-4}/plant-year and 10^{-2}/plant-year, and BDBEs with frequencies less than 10^{-4}/plant-year. The final category of LBEs, DBAs are derived from the DBEs using prescriptive rules to ensure that conservatively analyzed doses are within 10CFR50.34 dose limits without relying on any non-safety related SSCs for mitigation.
- Estimates of the frequencies and consequences of LBEs include mean values and uncertainty intervals that account for sources of uncertainty identified in the PRA.
- Many LBEs identify initiating events and event sequences that challenge the plant safety functions but result in successful termination with no release of radioactive material and, hence, no offsite exposures. Understanding the plant design features responsible for accident prevention is an essential outcome of the LBE process.
- Limitation of dose exposures for individual AOOs that may involve a release to 100mrem. For AOO frequencies of 1/per plant year and higher, an iso-risk profile is used ending at 100mrem at the 1/ per plant year frequency level. For AOO frequencies from 1/ plant-year to 10^{-2} per plant year, the AOO dose limit is fixed at 100mrem.
- The dose evaluation criteria for DBEs with a release range from 10% to 100% of the 10 CFR 50.34 dose limit of 25 rem for DBE frequencies that range from 10^{-2} to 10^{-4} per plant year, respectively.

- The dose evaluation criteria for BDBEs with a release range from 300 rem (low probability of early fatality) to 750 rem (high probability of early fatality) for BDBE frequencies that range from 10^{-4} per plant year to $5x10^{-7}$ per plant year, respectively.
- Though not an NRC regulatory requirement, this Figure also shows the EPA Protective Action Guideline limit for sheltering at 1 rem to reflect an NGNP user requirement for reducing the size of the Emergency Planning Zone.

The NUREG-1860 LBE selection approach defines similar frequency vs. consequence criteria, shown in Figure 3-2, to evaluate off normal event selection from a PRA. Both sets of frequency vs. consequence criteria are developed based on somewhat differing interpretations of the then existing U.S. regulatory requirements that include annual limits on the radiological doses from normal operation, radiological dose evaluation criteria for the evaluation of AOOs and postulated accidents, and NRC QHOs. Even though the two sets of criteria are based on the same underlying requirements, judgments are needed to associate the regulatory dose evaluation criteria to event frequencies because the U.S. regulatory requirements use qualitative statements in lieu of numerical frequency limits to describe the likelihoods of AOOs and postulated accidents to be evaluated against the dose limits. By moving toward a quantitative measure of likelihood, the approach can be described as performance based. With the benefit of an updated review of more recent regulatory bases, some refinements to the TLRC can be proposed.

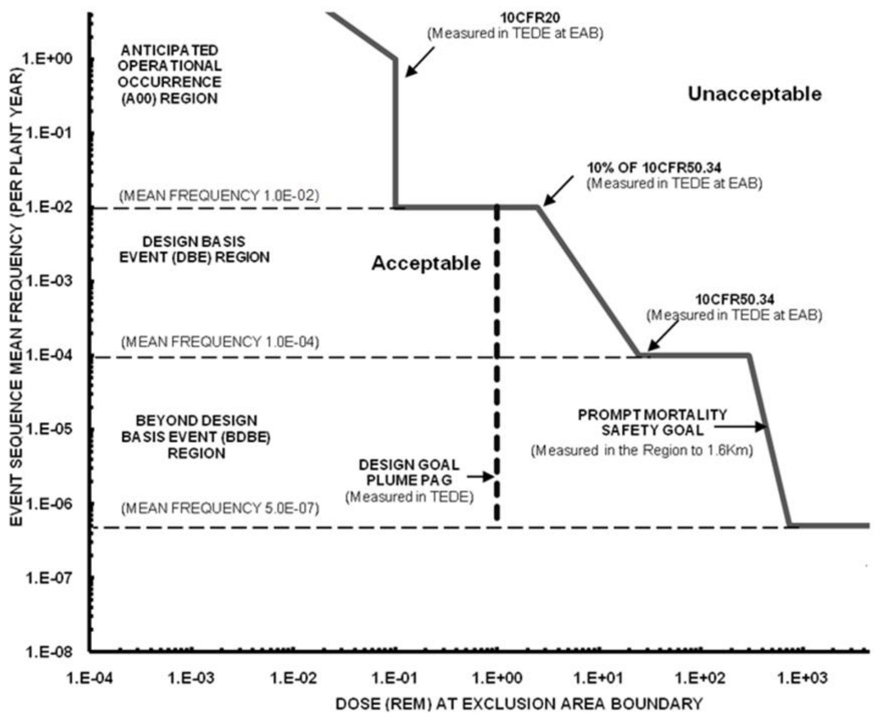

Figure 3-1 NGNP TLRC Frequency – Consequence Criteria

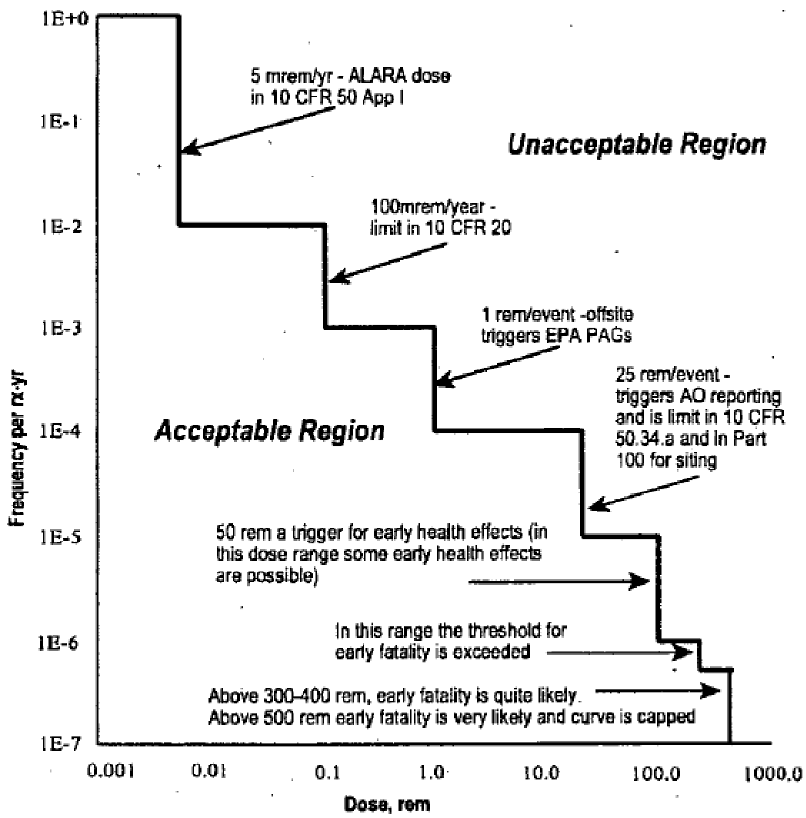

Figure 3-2 NUREG-1860 Frequency – Consequence Criteria

The review of the NGNP and NUREG-1860 approaches for defining the TLRC frequency-consequence criteria has identified three areas for improvement. These areas include a new insight into the interpretation of the 10 CFR 20 requirements for application to evaluating AOOs, a practical concern regarding the use of a "staircase" shape for the frequency-dose profile, and the consideration of risk aversion in setting the frequency vs. dose evaluation boundaries for the AOO, DBE, and BDBE categories of LBEs.

3.2.1 Interpretation of 10 CFR 20 and 10 CFR 50 Annual Exposure Limits

Both the NGNP and NUREG-1860 approaches defined TLRC frequency vs. dose criteria for evaluating the frequencies and doses of individual LBEs. NGNP used 10 CFR 20 to define an isorisk line (i.e., line of constant risk defined as the product of the frequency and dose) for frequencies greater than 1/plant year, but as a fixed dose limit for events with frequencies between 1 and 10^{-2} per plant-year. NUREG-1860 used the annual dose limits of 10 CFR 50 to limit doses from individual event sequences with frequencies between 1 and 10^{-2} per reactor-year, and the annual dose limits of 10 CFR 20 to limit doses of individual LBEs at frequencies between 10^{-2} and 10^{-3} per reactor-year. As noted in SRP Chapter 15.0, the doses from AOOs having a relatively low frequency of occurrence may exceed 10 CFR 20 so long as the risk, defined by the product of the frequency and consequence, is sufficiently low and other limits are not exceeded.

> *"If the risk of an event is defined as the product of the event's frequency of occurrence and its consequences, then the design of the plant should be such that all the AOOs and postulated accidents produce about the same level of risk (i.e., the risk is approximately constant across the spectrum of AOOs and postulated accidents). This is reflected in the general design criteria (GDC), which generally prohibit relatively frequent events (AOOs) from resulting in serious*

consequences, but allow the relatively rare events (postulated accidents) to produce more severe consequences."

However according to SRP Chapter 15.0 the doses of lowest frequency AOOs for PWRs:

> "*..shall not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius"*

One metric that could be used to demonstrate that doses are not sufficient to restrict use beyond the EAB is the 1 rem EPA PAG limit for initiating offsite protective actions. This same part of the SRP states that for lower frequency AOOs for BWRs:

> *"..the offsite release of radioactive material is limited to a small fraction of the guidelines of 10 CFR Part 100."*

There are several places in the SRP, such as Chapter 15.0.3 where the concept of a "small fraction" is interpreted as 10%:

> *"A small fraction is defined as less than 10% of the 10 CFR 50.34(a)(1) reference values, or 2.5 rem TEDE."*

2.5 rem is 10% of the 10 CFR 50.34 dose limit of 25rem. From these acceptance criteria in Chapter 15 of the SRP it is reasonable to permit the doses from the lower frequency AOOs to be as high as 1 rem (EPA PAG limit) to 2.5 rem (small fraction of 10 CFR 50.34 limit) for consistency with LWR AOO acceptance criteria.

The above statements from SRP Chapter 15 suggest that both the NGNP and NUREG-1860 frequency-consequence criteria are too conservative in interpreting the 10 CFR 20 annual dose limits as not to exceed criteria for individual LBEs. This insight is used to propose alternative frequency-consequence criteria in Section 3.3.

3.2.2 "Staircase Discontinuity Issue

A second issue that was identified from reviewing the NGNP and NUREG-1860[9] frequency vs. consequence criteria is referred to here as the "staircase discontinuity" issue. This issue is illustrated in Figure 3-3 using the NGNP criteria as an example, however both sets of criteria are subject to this issue. The NUREG-1860 version of the criteria is essentially a succession of smaller staircase steps. The NGNP criteria are based on an interpretation that 10 CFR 20 annual dose limit should be used as limit on the dose from individual AOOs with frequencies from 1 per plant-year to 10^{-2} per plant-year. This lower AOO region frequency of 10^{-2} per plant year is also the upper limit of the DBE region in the NGNP criteria. The NGNP dose criteria for the DBE region range from 10% of the 10 CFR 50.34 limit at the top end of the frequency range to 100% of the same dose limit at the bottom end of the DBE frequency range of 10^{-4} per plant year.

There is another staircase in the transition from the DBE region to the BDBE region in the NGNP criteria. The lower limit of the BDBE region is set at a frequency ($5x10^{-7}$/plant-year) at which it can be assured that the QHOs for early health effects can be satisfied independent of the level of consequences. The dose assigned to the criteria at this frequency is 750rem TEDE, which is associated with a high probability of death for a person located at the EAB. The dose assignment at the $5x10^{-7}$ frequency level is already a very conservative representation of the QHOs because the early fatality QHO is based on the average individual risk over the entire area between the EAB and 1 mile out from the EAB. Considering the fact that the dose drops off with distance across the 1 mile "doughnut" and also drops off very rapidly

[9] The risk targets in the UK SAPs also obey a "staircase" shape when plotted on a log frequency vs. log consequence graph.

as one moves off the center-line of the direction of the release plume, the average individual risk would be much less than the QHO for an accident at much higher frequencies than $5x10^{-7}$ even if the EAB dose at the center line of plume would approach 750rem. A step is included in the NGNP approach to confirm that QHOs have been met based on an integrated assessment of the individual risk in the area surrounding the plant accounting for all the LBEs.

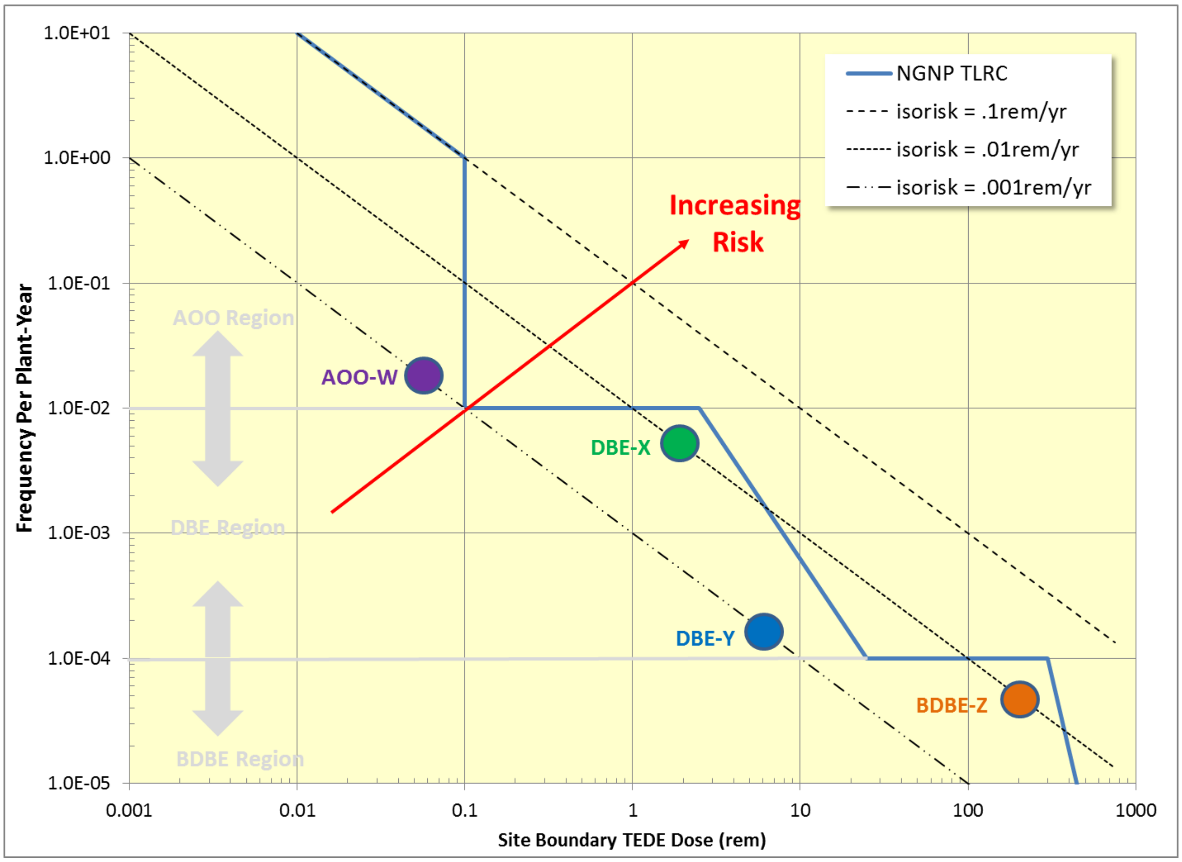

Figure 3-3 Part of NGNP Frequency-Consequence Criteria Illustrating Staircase Issue

The combination of assumptions used to draw the NGNP frequency vs. consequence curve from 1 per plant year to 10^{-2} per plant year creates an undesirable property that the "risk" as defined as the product of the frequency and dose is allowed to increase as the LBE frequencies change from the lower end of the AOO region in the vicinity of AOO labeled AOO-W to the upper end of the DBE region in the vicinity of DBE-X. In these examples the risk of DBE-X is an order of magnitude greater than that for AOO-W. There is a similar problem when comparing the risks between DBE-Y and BDBE-Z in which case BDBE-Z, having a somewhat smaller frequency has a higher risk than DBE-Y, while both are on the acceptable side of the frequency-consequence criteria. In both of these examples the frequency-consequence criteria permit higher risks with small reductions in frequency at the transitions across the staircase steps. NUREG-1860 also has these staircase discontinuities; however there are more steps in those frequency-consequence criteria which somewhat alleviate the concern.

This staircase issue creates an implementation issue to be avoided in designing and licensing a new reactor. As the design matures and the PRA is updated to incorporate plant design changes and refinements to requirements, the LBE frequencies and consequences are subject to and will likely change. In addition, the LBE frequency and consequence estimates are not points but cover a range of values within the uncertainty bounds which may overlap the frequency cut-offs for each LBE region. It is problematic to permit large changes in allowable risk when transitioning from one LBE region to another which any staircase criteria would suggest. The uncertainties in frequency and consequence estimates as well as the changes in the estimates that would be expected from successive PRA updates during the evolution of design require a more continuous behavior in the risk acceptance criteria.

The NGNP frequency-consequence criteria has one "knee" in the curve at the upper end of the BDBE region established by the upper frequency limit of 10^{-4}/plant year and a dose limit of 300rem, which corresponds to a probability of fatality due to prompt radiation syndrome of .005, i.e. the lower bound of the probability of death vs. exposure curve. Using 25rem at the lower end of the DBE region and 300rem where the BDBE region starts and DBE region leaves off is a big increase and requires a good justification. There does not exist a regulatory basis for this point at the knee. The derivation of this knee appears to be based on a mixture of accident frequencies and conditional probability of a certain consequence from a release. The probability of fatality from a given exposure is an important factor to consider in estimating the individual risk of fatality from a given release. However the **probability** of fatality curve has been used here as a basis for establishing a relationship between event **frequencies** and acceptable exposures for events at a given frequency. This then yields a risk, defined as the product of the frequency and dose at the upper end of the BDBE region, which is much higher than that at the lower end of the BDBE region and at the lower end of the AOO region. Hence it does not appear that a good basis exists for establishing the upper knee of the criteria in the BDBE region at 300 rem and 10^{-4}.

To avoid these issues, it is desirable that the acceptable risk levels should not be allowed to increase when transitioning from the AOO region to the DBE region, or when transitioning from the DBE region to the BDBE region. This consideration leads to some proposed refinements to the frequency-consequence criteria that are proposed for the LMP project as described in Section 3.3.

3.2.3 Plant-year vs. Reactor-year Frequency Basis

A key difference between the NGNP and NUREG-1860 frequency-consequence criteria is the different frequency bases that are used. NGNP defines the frequency basis on a per plant-year basis where a plant may be comprised of two or more reactor modules in order to address LBEs that may involve releases from two or more reactor modules or sources of radioactive material. This approach addresses the increased likelihood in the frequency of single-unit events that occur on each reactor modules independently and enables a meaningful comparison of the frequencies of single and multi-module events as well as events from a common radionuclide source such as fuel storage facility. NUREG-1860 retains the traditional PRA approach that has been used for operating LWRs where LBEs derived from PRAs are addressed for each reactor on one-reactor-at-a-time basis, which leads to expressing frequencies on a per reactor-year basis. This NUREG-1860 approach makes it problematic to compare LBEs that involve single and multiple reactor source terms and fails to measure the increased likelihood of independent events occurring on each module independently. Lessons learned from the Fukushima Daiichi accident as exemplified in the NTTF report cannot be effectively addressed by an evaluation that is done on a one-reactor-at-a-time basis. The need to address both single reactor and multiple reactor events was highlighted in several other regulatory precedents that were reviewed in Section 2 including SRP Chapter 19, SECY-2003-0047, and several other references. For example, it is noted that the UK SAPs include frequency-consequence criteria both on a per reactor-year basis, for the purpose of evaluating the generic design assessments similar to the U.S. design certifications, and on a per site-year basis for addressing the integrated risks for an entire site. The LMP project prefers the NGNP approach because it provides a basis to address LBEs for a multi-module plant design. Addressing the integrated risks of an entire site, which may include other plants not within the scope of an advanced non-LWR license application, is considered beyond the scope of this project. Hence the per plant-year frequency basis is selected for use in the LMP frequency-consequence criteria.

3.2.4 Risk Aversion Considerations

The next topic addressed in the review of proposed frequency vs. consequence criteria is the principle of risk aversion. In application of this principle, risk targets of low frequency events which may have large consequences are set to lower criteria of risk than those for the higher frequency events which are expected to have lower consequences. One of the first proposals for numerical frequency vs. consequence criteria was made by Reginald Farmer of the United Kingdom, who is also recognized as the father of PRAs applied to assess the risks of reactor accidents [64]. His proposed limit lines are illustrated in Figure 3-4 and are expressed in terms of accident frequency vs. quantity of release of the key radionuclide I-131. This risk metric was used in the early days of nuclear power in the U.K. to address the question "How safe is safe enough"? The top limit line follows an isorisk contour at frequencies below 10^{-3} per year, which when plotted on log-log paper is shown as a straight line with logarithmic slope of -1. The iso-risk contour is neutral with respect to the risk aversion principle. The principle of risk aversion is applied in the lower two curves which have steeper logarithmic slopes of -1.33 and -1.5, respectively at frequencies below about 10^{-3}/year. The curve having the greatest allowance for risk aversion is the lowest curve with logarithmic slope of -1.5, which is the acceptance criterion proposed by Farmer.

The principle of risk aversion is considered in the formulation of revised frequency –consequence criteria as developed in Section 3.3.

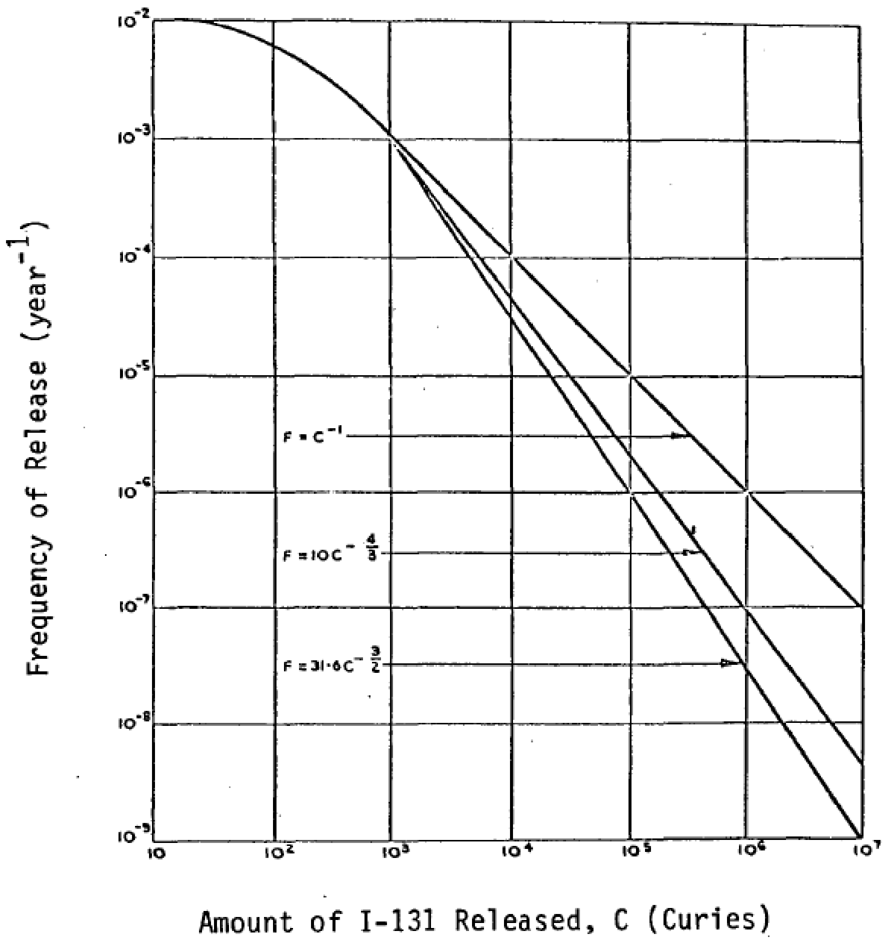

Figure 3-4 Frequency vs. Consequence Limit Line Proposed by Farmer [64]

3.2.5 Definition of LBE Categories

The NGNP approach for definition of LBE frequency categories of AOO, DBE, and BDBE is somewhat different than the approach used in NUREG-1860. The latter reference uses the traditional PRA frequency metric of events/reactor-year whereas NGNP uses frequency/plant-year where a plant may be comprised of multiple reactor modules as discussed in Reference [1]. NUREG-1860 classifies events as frequent with frequencies greater than 10^{-2}/reactor-year, infrequent with frequencies between 10^{-5}/reactor-year and 10^{-2}/reactor-year, and rare with frequencies less than 10^{-5}/reactor-year. NUREG-1860 does not address the multi-module risk issue. However, considering that modular reactor plants with as many as 12 modules have been proposed, the frequency classes proposed in the respective references are comparable, as will be shown in the next Section. As noted in ANS Standard 53.1, "Nuclear Safety Design Process for the Design of Modular Helium-Cooled Reactor Plants"[25].

> *"The adoption of the DBE region's lower frequency limit of 1 × 10^{-4} (1E-04) per plant-year is appropriate because it is applied on a per-plant basis and accounts for possible multiple MHR modules. In addition, the expression of the frequency metric on a per plant-year basis enables the assessment to include event sequences involving only one or multiple reactor module source terms and thereby provides a more complete risk assessment as compared with the approach of analyzing each reactor module on an independent reactor-year basis."*

As noted in the SRM to SECY- 90-16 [35], the Commission endorsed a core damage frequency (CDF) goal of 10^{-4} per reactor year for advanced reactors. However, even for advanced reactors, core damage events are regarded as beyond design basis accidents. Hence, the selection of a lower limit of the DBE region of 10^{-4}/plant-year, where a plant may be comprised of multiple reactor modules, is conservative relative to the Commission's advanced reactor CDF goal, which is only for a single reactor.

The LMP project has chosen the NGNP LBE frequency criteria for differentiating between AOOs, DBEs, and BDBEs. This selection is made in order to support the capability to address multi-module events and events that may involve two or more sources of radioactive material.

3.2.6 Risk Evaluation of LBEs and Integrated Risk Assessment

The approach to evaluating risk significance of LBEs in the NGNP approach is done in two levels. The estimates of frequencies and site boundary doses, including their mean values and uncertainty distribution upper and lower percentiles are evaluated for each LBE individually against the TLRC frequency-consequence criteria to determine the appropriate LBE category of AOO, DBE, and BDBE.

Consistent with the NGNP approach, in classifying individual LBEs into the correct bins based on frequency in the LMP approach, when the mean frequencies are in the AOO region but the lower 5% of the frequency is in the DBE region, the LBE shall evaluated both as an AOO and a DBE. If the mean frequency is less than 10^{-4}/plant-year and the upper 95%tile frequency is in the DBE region, the LBE is also evaluated as a DBE. Hence, when the uncertainty band on the frequencies straddles an LBE frequency criterion that separates AOOs, from DBEs, and BDBEs, the LBE is evaluated using the criteria in both regions.

Second, in addition to the risk evaluation of individual LBEs, the integrated risks considering the total from all the LBEs is also considered in the NGNP approach. For this purpose, the risk metrics to be used are those for comparison against the two NRC QHOs. The QHOs are defined in the NRC Safety Goal Policy Statement. As noted in the previous section, that policy statement also has a performance goal for maintaining the frequency of a large release below 10^{-6}/year. It is recognized that there has been some differing views as to whether the NRC safety goals and the associated QHOs and associated performance goals should be applied on a per site-year, or reactor-year basis. However, it is reasonable to interpret LRF as a reactor technology-neutral performance goal. Hence, the LMP approach includes an additional LRF goal as well as the two QHOs. In the PRISM PRAs the LRF goal was conservatively interpreted as a goal to prevent the frequency of exceeding 25rem at the site boundary to be less than 10^{-6} per reactor year for a two-reactor module plant[69]. As discussed more fully in SECY 2013-0029 [71] the use of 25 rem as a conservative definition of large early release has been used in the PRAs developed for several ALWR design certifications, including the ESBWR based on an EPRI report on user requirements for advanced reactors. This definition a large early release is viewed as too conservative because a large release from an LWR for which the policy is applied to[10] would be expected to have the potential to produce site boundary doses much larger than 25rem.

A dose limit of 750rem is selected to define the large early release goal for the LMP approach. This is consistent with the way in which the individual risk QHO equivalent dose was assigned in the definition of the TLRC frequency-dose evaluation criteria. This is still viewed as conservative because this is the dose on the centerline of the plume using conservative meteorology assumptions at the site boundary which would essentially guarantee very few early fatalities, if any, in the 1 mile area beyond the site boundary. Stating the goal in to limit the frequency of exceeding 750rem to less than 10^{-6} per plant year can be viewed as a surrogate to the individual risk QHO. By using this as a surrogate for the early fatality QHO, it is possible for an LWR to utilize a Level 2 PRA and a simple dose calculation to demonstrate compliance with the early fatality QHO without having to perform a Level 3 PRA which is not required for ALWR design certifications.

As noted in Section 3.2.1, the annual dose limits in 10 CFR 20 have been used to derive TLRC frequency-dose criteria for evaluating individual LBEs in the AOO region. These annual dose limits also imply a limit on the integrated risks of LBEs that supplement those derived from the Safety Goal Policy and serve to limit the cumulative risks of LBEs in the high frequency and low consequence part of the risk spectrum. Although it is reasonable to use 10CFR20 annual dose limits as a basis for deriving frequency-dose criteria for higher frequency LBEs, as has been done for both NGNP and NUREG-1860, just meeting the criteria for events evaluated individually does not necessary satisfy the integrated annual doses that are limited in 10 CFR 20. Hence this additional integrated risk metric has been added to confirm that 10 CFR 20 is met for the summation of all the LBEs.

In view of these considerations the LMP has chosen to retain the two QHO goals used in the NGNP approach for evaluating the integrated risks and add two additional goals to address the LRF goal and to address the 10 CFR 20 annual dose limits:

- The total frequency of exceeding a site boundary dose of 100 mrem shall not exceed 1/plant-year to ensure that the annual exposure limits in 10 CFR 20 are not exceeded.
- The total frequency of a site boundary dose exceeding 750 rem shall not exceed 10^{-6}/plant-year to address the LRF goal in the Safety Goal Policy and also to maintain the frequency of any and all accidents with dose consequences exceeding the limit for DBAs acceptably small and consistent with expectations for advanced non-LWRs.
- The average individual risk of early fatality within the area 1 mile of the EAB shall not exceed $5x10^{-7}$/plant-year to ensure that the NRC Safety Goal QHO for early fatality risk is met

[10] The term large release is generally used to mean a release that has the potential to cause life threatening radiation exposures off-site. As noted in Figure 3-2, NUREG-1860 states that the first signs of early health effects require the dose to exceed 50rem.

- The average individual risk of latent cancer fatalities within the area 10 miles of the EAB shall not exceed $2x10^{-6}$/plant-year to ensure that the NRC safety goal QHO for latent cancer fatality risk is met.

It is noted that the above integrated risk goals are in addition to the requirements to maintain the risks for the individual LBEs within the TLRC frequency-dose criteria, and are applied on a per plant-year basis so that the integrated risks of accidents involving single and multiple reactor modules and sources of radioactive material are included. The application of these integrated risk criteria to a plant which may be comprised of multiple reactor modules, rather than to individual reactors is consistent with an ACRS recommendation on treatment of integrated risks for advanced non-LWR designs [70], which states:

> *The Quantitative Health Objectives (QHOs) apply to the site as a whole. The sum of the contributions from each reactor on the site to acute and latent fatalities should be bounded by the QHOs.*

3.2.7 Summary of Review Findings

The take-away lessons from this part of the LBE approach review are as follows:

- Both the NUREG-1860 and NGNP approaches to LBE selection appear to meet the LBE selection attributes listed in Section 3.1. The major elements of both approaches are comparable and are judged to be capable of providing the desired characteristics of being risk-informed, performance-based, reproducible, and capable of identifying a sufficiently complete set of limiting reactor specific LBEs.
- The interpretation of 10 CFR 20 annual exposure criteria as a risk limit, rather than a dose limit for individual events is more consistent with the intent of the current regulatory requirements as discussed in SRP Chapter 15.0.
- For consistency with requirements for lower frequency AOO events in SRP 15.0, which state that doses from lower frequency AOOs should not impact offsite activities, the radiological exposures for lower frequency AOOs should not exceed the EPA PAG limits for triggering offsite protective actions. Limiting lower frequency AOO events to 1rem TEDE at the EAB would satisfy this requirement.
- It is undesirable to have staircase discontinuities in the frequency vs. dose criteria. Otherwise, small reductions in LBE frequencies would yield higher acceptable risks as the frequency thresholds at the stair steps are crossed. Small changes in LBE frequencies in any part of the frequency-consequence spectrum should be evaluated against small changes in dose criteria, and not against large step changes in criteria.
- In order to apply the principle of risk aversion it is appropriate to accept lower criteria on risks for lower frequency accidents in the DBE and BDBE regions which have the potential for higher consequences.
- The NGNP approach to defining LBE frequency categories of AOO, DBE, and BDBE on the basis of frequency per plant-year is preferred over the NUREG-1860 method which uses a per reactor-year frequency basis due to its superior capability to address multi-module events.

These lessons are reflected in proposed frequency vs. consequence criteria as described in the next Section.

3.3 Proposed Revisions to NGNP TLRC Frequency – Consequence Evaluation Criteria

Based on insights from the review of existing criteria, the LMP proposes to use a set of frequency – consequence criteria that adopt the NGNP criteria as a basis with some refinements to address the review insights from the previous section. The criteria proposed in NUREG-1860 are used as guidance and as a sanity check to compare against the proposed criteria after some adjustments that attempt to reconcile the differences in frequency bases. The reason for starting with the NGNP approach is that this approach is more consistent with LMP project objectives. The following elements of the NGNP approach are viewed as more consistent with the LBE attributes discussed in Section 3.1, namely:

- The classification of LBEs into distinct categories of AOOs, DBEs, BDBEs that lead to systematic identification of DBAs provides a clean interface with the results of the PRA and prescriptive inputs to the deterministic safety analysis. This approach addresses the needs of the designer and supports the attribute of predictability of the licensing process.
- The NGNP approach to selection of LBEs has its origins in the MHTGR design and licensing approach. It provides benefits to the full plant application of RIPB insights that supports LBE selection, SSC safety classification, and derivation of reactor specific Principle Design Criteria.
- The frequency basis of events/plant-year facilitates application to advanced non-LWR plant designs using a modular reactor approach. This provides the designer with capabilities to define LBEs and to develop design strategies to prevent and mitigate accidents involving multiple reactor modules and non-core radionuclide sources.

The TLRC frequency –consequence evaluation criteria proposed for the LMP project are shown in Figure 3-5. These criteria are based on the following considerations:

- The regions of the graph separated by the frequency-dose evaluation line are identified as "risk significant" rather than "unacceptable", and "risk insignificant" rather than "acceptable" to emphasize that the purpose of criteria is to evaluate the risk significance of individual LBEs and to recognize that risk evaluations are not performed on a pass-fail basis in contrast with deterministic safety evaluation criteria. This change is consistent with NRC risk-informed policies such as those expressed in RG. 1.174 in which risk is not "accepted" but rather evaluated for risk significance.
- The evaluation line doses for high frequency AOOs down to a frequency of 10^{-1}/plant-year are based on an iso-risk profile defined by the annual exposure limits of 10 CFR 20, or 100rem/plant-year.
- The doses for AOOs at frequencies less than 10^{-1}/plant-year are capped at 1rem corresponding with the EPA PAG limits and consistent with SRP Chapter 15.0 acceptance criteria for lower frequency AOOs for PWRs.
- The dose criteria for DBEs range from 1rem at 10^{-2}/plant-year to 25rem at 10^{-4}/plant-year. Considering this is a frequency per plant-year which may be applied to multi-module plants, the 10^{-4} target for the bottom of the DBE range is more conservative than the NRC CDF goal for advanced reactors at 10^{-4}/reactor-year [35]. This limits the lowest frequency DBEs to the limits in 10 CFR 50.34 and provides continuity to the lower end of the AOO criteria. A straight line on the log-log plot connects these criteria.
- The dose criteria for the BDBEs range from 25 rem at 10^{-4}/plant-year to 750 rem at 1×10^{-6}/plant year to meet the anchor point for the LRF goal. The BDBE doses from 1×10^{-6}/plant year down to 5×10^{-7}/plant-year are fixed at 750 rem, providing continuity with the lower end of the DBE region and using the same conservative interpretation of the early health effects QHO as used to anchor this end of the NGNP criteria.

- The frequency-dose anchor points used to define the shape of the curve are indicated in Figure 3-5.
- In consideration of the risk aversion principle, the logarithmic slope of the curve in the DBE and BDBE regions exceeds -1.5 which corresponds to the most conservative limit-line proposed by Farmer to address risk aversion as shown in Figure 3-4.

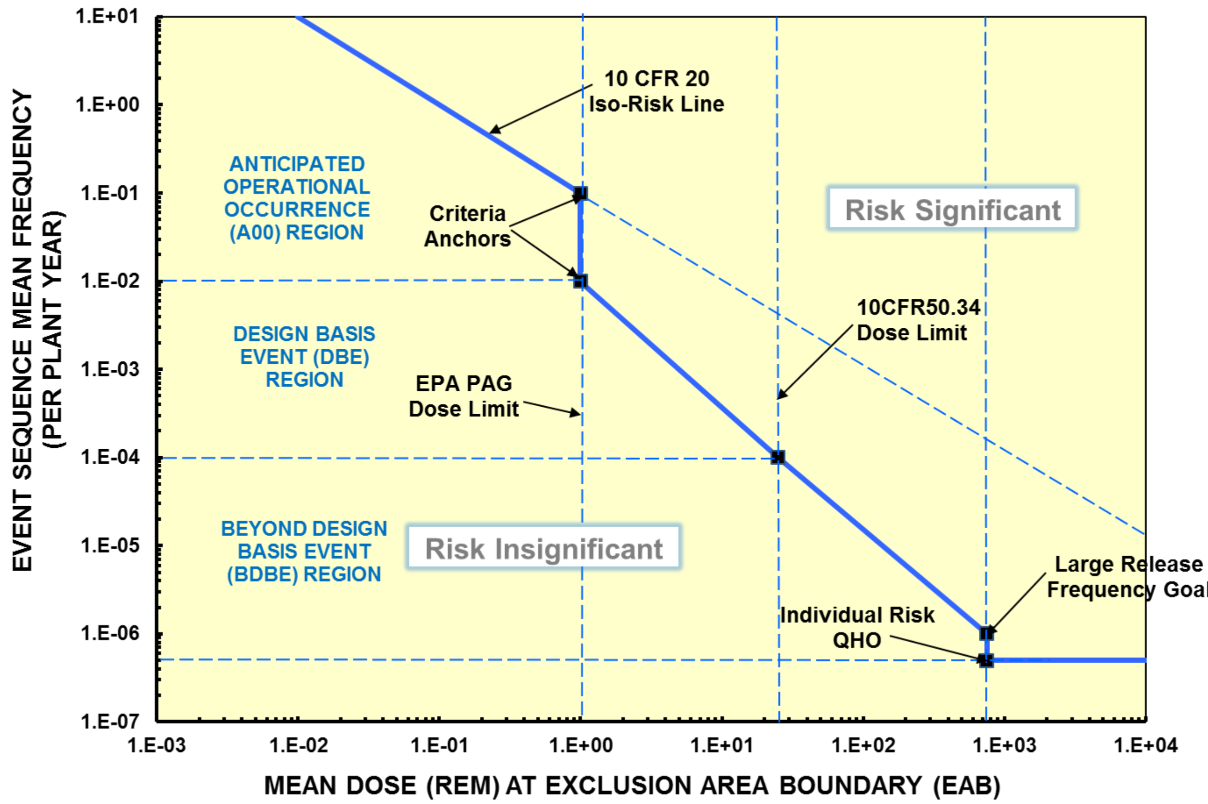

Figure 3-5 Frequency-Consequence Evaluation Criteria Proposed for LMP

The criteria in Figure 3-5 address the issues raised in the criteria review discussed in the previous section. This formulation eliminates the staircase issues. Across the entire spectrum the risk defined as the product of the frequency and consequence is not permitted to increase as the frequency decreases. In addition, the principle of risk aversion is applied at frequencies below 10^{-1}/plant-year. The logarithmic slope of the criteria between 10^{-2}/plant-year to $5x10^{-7}$/plant-year is about -1.5 consistent with Farmer's limit line.

While interpreting the 10 CFR 20 annual exposure limits of 100 mrem/year, it is recognized that the proposed use of this criteria is to be applied to individual LBEs. In order to ensure that the cumulative releases considering all the LBEs do not exceed this limit, the LMP LBE proposes to add a task not included in the NGNP LBE to insure that the integrated risks summed over all the LBEs do not exceed 100 mrem/year. The proposed LBE approach also retains the NGNP task of performing an integrated assessment over all the LBEs to ensure that NRC safety goal QHOs for both early and latent health effects are met.

A comparison of the proposed criteria and the NGNP criteria is shown in Figure 3-6. As seen in this figure, the LMP proposed criteria are less restrictive in the AOO region taking advantage of more up to date interpretations of the existing SRP acceptance criteria for AOOs in LWRs. However the proposed criteria are somewhat more restrictive in the DBE and BDBE regions. Based on examples from HTGR and sodium fast reactor PRAs presented in this report and in a companion paper on PRA development, it is expected that this will not cause any issues with advanced reactor designs. This modification primarily results from resolving the "staircase" discontinuity issues with the NGNP criteria.

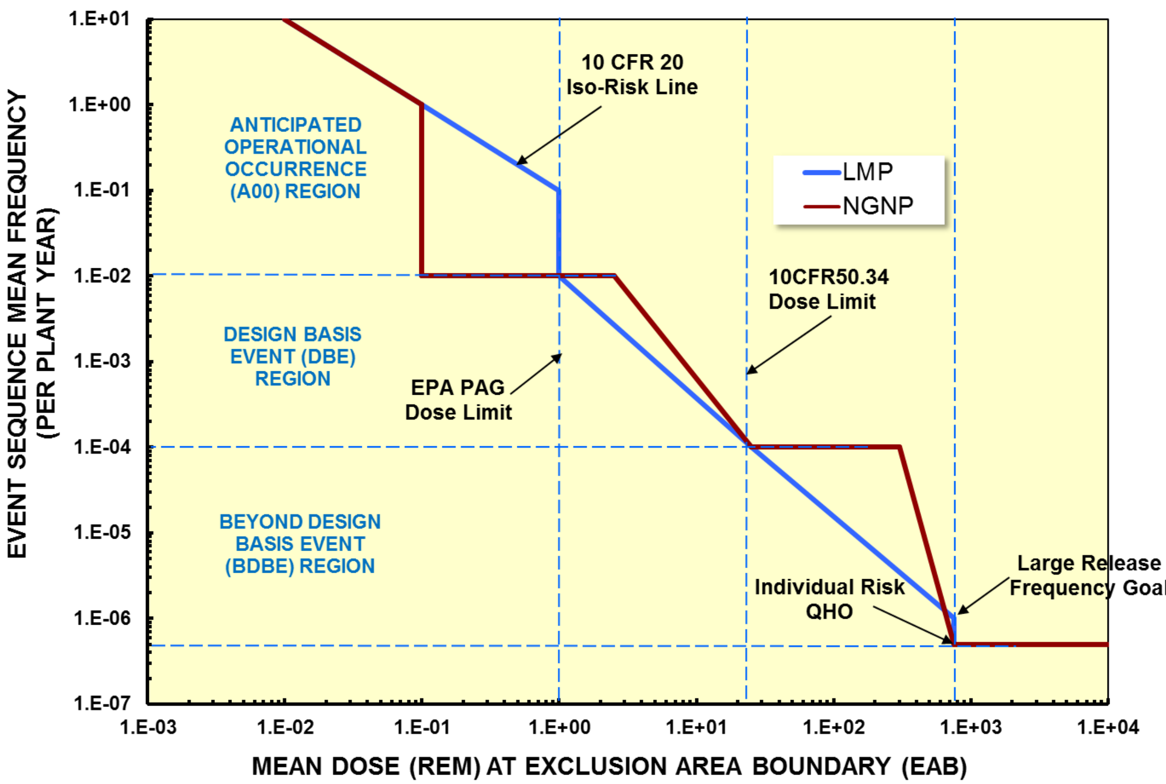

Figure 3-6 Comparison of LMP and NGNP Frequency – Consequence Criteria

A similar comparison is made in Figure 3-7, in this case to contrast the LMP and the NUREG-1860 criteria. The NUREG-1860 criteria are expressed on a per reactor-year basis. In order to compare against the LMP criteria, which are for a multi-module plant, three versions of the NUREG-1860 are shown, one for a 1-module plant, one for a 4-module plant, and a third for a 12-module plant. The 4-module and 12-module versions are obtained by simply scaling the frequencies which is recognized to be appropriate only for event sequences affecting a single module. As seen in this comparison, the NUREG-1860 criteria are much more restrictive for high frequency events with frequencies above 10^{-2}/plant-year. However for frequencies below about 10^{-3}/plant-year the respective criteria are quite comparable. The LMP criteria for the high-frequency range are judged to be more consistent with the most recent update of the SRP Chapter 15 for criteria used to evaluate AOOs in current generation LWRs. The NUREG-1860 criteria are based in part on the use of annual exposure limits from 10 CFR 50 Appendix I. However such criteria are not used in the SRP Chapter 15.0 for evaluating exposures for individual AOOs for LWRs. For the lower frequency range, the primary difference between the two sets of criteria is seen to be due to the staircase effects. On balance, these comparisons provide a useful sanity check on the reasonableness of the proposed criteria for the frequencies and consequences of LBEs for the LMP project.

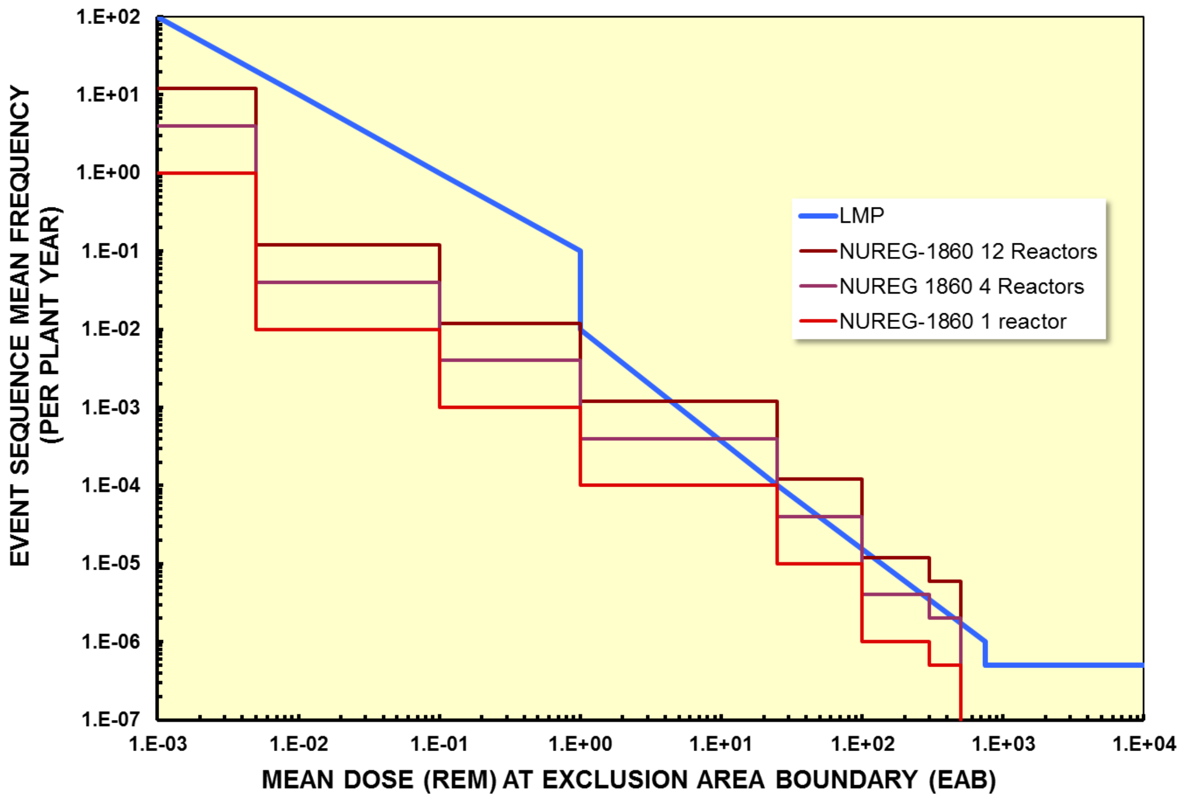

Figure 3-7 Comparison of LMP and NUREG-1860 Frequency – Consequence Criteria

3.4 LMP LBE Selection Process

3.4.1 LBE Selection Process Overview

The design of advanced non-LWRs will be developed using a systematic, top-down TI-RIPB approach to meeting regulatory and end-user requirements that achieves the LBE selection attributes listed in Section 3.1. Appropriate engineering design and analysis techniques will be used to make design selections to satisfy these requirements. Regulatory requirements must include nuclear safety considerations to protect the offsite public and onsite workers from radioactive materials. The design cannot be advanced beyond that to meet the end user requirements without establishing the safety design approach to be implemented in the design. The safety design approach includes the selection of materials and design features for the reactor components, characterization of the sources of radioactive material, selection and arrangement of radionuclide transport barriers, definition of safety functions to protect these barriers, and selection of SSCs for the performance of these and other energy production functions. The safety design approach must anticipate the challenges to the safety case[11] that include those unique and specific to the reactor technology. Hence, it is necessary for the designer to perform an early assessment to identify the LBEs that frame the necessary safety analyses that will be performed to demonstrate adequacy of the safety case. These analyses include those necessary and sufficient to confirm the adequacy of defense-in-depth in the prevention and mitigation of accidents. Given these considerations, it is clear that the selection of

---

[11] As the term is used in this document, the term "safety case" is the collection of statements about the capabilities of the reactor design and intended means of operation, that if demonstrated to be true, would ensure an adequate level of safety to protect the public.

LBEs must begin early in the design process in order to optimize the design in meeting end user and regulatory requirements and to avoid costly back-fits that might otherwise occur during late stages of design and licensing.

A flow chart indicating the steps to identify and evaluate LBEs in concert with the design evolution is shown in Figure 3-8. These steps are intended to be carried out by the design and design evaluation teams responsible for establishing the key elements of the safety case and preparing a license application. The process is used to prepare an appropriate licensing document, e.g., licensing topical report, that documents the derivation of the LBEs, which would be reviewed by the regulator as part of license review. The design and design evaluation teams are responsible for selecting the LBEs and the regulator is responsible to review and approve the selections as well as the process used for the selections. Although it is anticipated that NRC would review the entire LBE selection and evaluation process, the specific steps with increased regulatory involvement are identified in the figure. The LBE selection and evaluation process is implemented in the following LBE selection tasks:

Task 1 Propose Initial List of LBEs
In order to begin the design, it is necessary to select an initial set of LBEs which may not be complete but is necessary to develop the basic elements of the safety design approach. These events are selected deterministically based on all relevant and available experience including experience from the design and licensing of reactors of a different technology. In many cases, the designer may also have an initial assessment regarding which SSCs will be classified as safety related to meet the business case for the reactor design. This classification would also be deterministically based using the same information utilized for the initial selection of LBEs.

Task 2 Design Development and Analysis
The design development is performed in phases and often includes a pre-conceptual, conceptual, preliminary, and final design phase and may include iterations within phases. The subsequent Tasks 3 through 9 are repeated for each design phase until the list of LBEs is finalized. Because the selection of deterministic DBAs requires the selection of safety related SSCs, this process also yields the selection of safety related SSCs that will be needed for the deterministic safety analysis in Task 7a. The sequence of design phases would be somewhat different if the LBEs are being used to support a Design Certification Application or a Combined Operating License.

Task 3 PRA Development/Update
A PRA model is developed and updated for each phase of the design. In the first design phase, which is typically the pre-conceptual design, the PRA is of limited scope and coarse level of detail and makes use of engineering judgment much more than a completed PRA that would meet applicable PRA standards [32]. The scope and level of detail of the PRA are then enhanced as the design matures and siting information is defined. More information on the PRA approach to support LBE development is the topic of a companion paper on the LMP approach to PRA. For modular reactor designs, the event sequences modeled in the PRA would include event sequences involving a single or multiple reactor modules. This approach provides the necessary risk insights to the design to ensure that accident sequences involving multiple reactor modules are not risk significant. The PRA provides estimates of the frequencies and doses for each LBE including a quantification of the impacts of uncertainties.

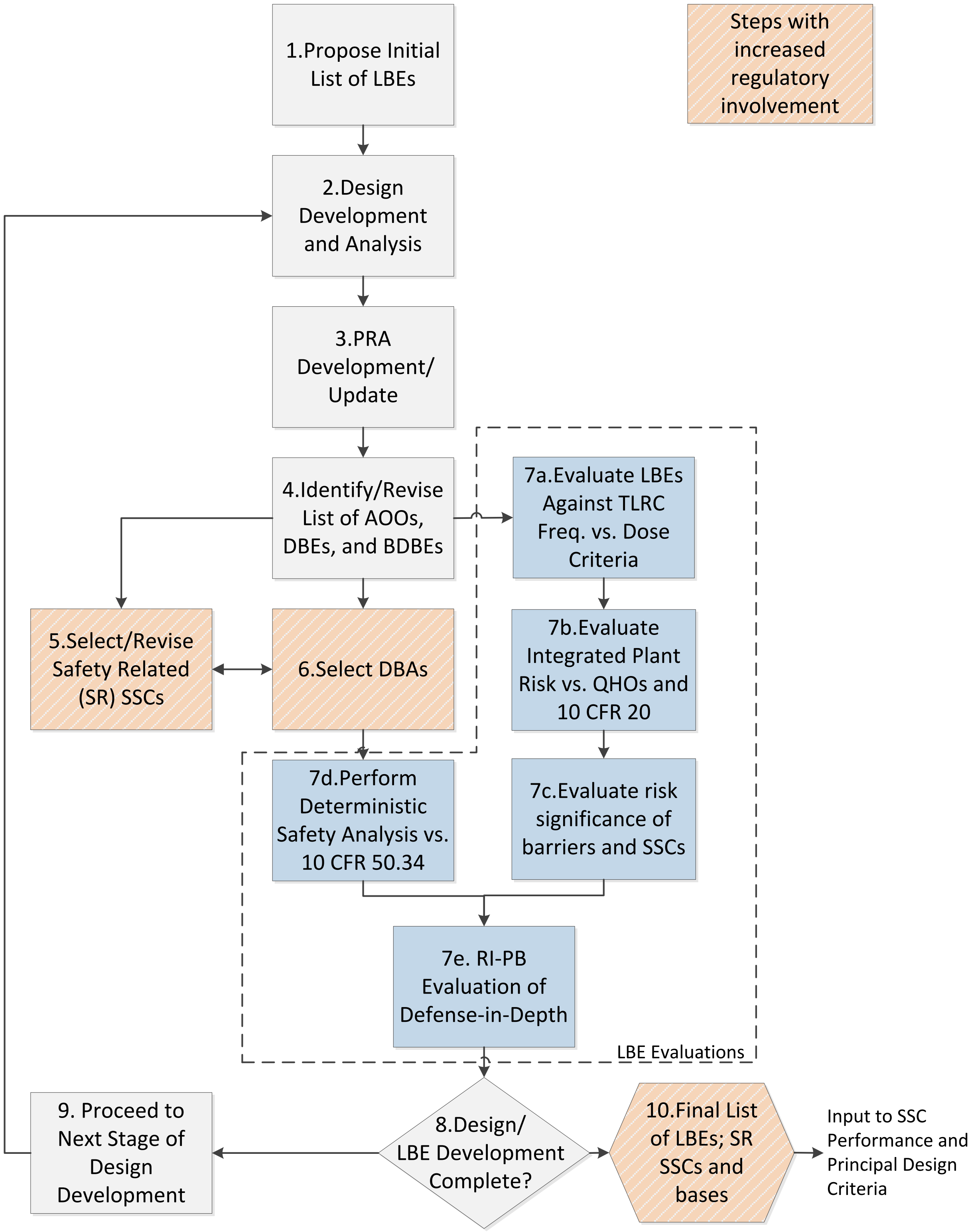

Figure 3-8 Process For Selecting and Evaluating Licensing Basis Events

Task 4 Identify/Revise List of AOOs, DBEs, and BDBEs
The event sequences modeled and evaluated in the PRA are grouped into accident families each having a similar initiating event, challenge to the plant safety functions, plant response, and mechanistic source term if there is a release. Each of these families is assigned to an LBE category based on mean event sequence frequency of occurrence per plant-year. AOOs have mean frequencies greater than 10^{-2}/plant-year; DBEs have mean frequencies between 10^{-4} and 10^{-2}/plant-year; and BDBEs have mean frequencies between $5x10^{-7}$/plant-year and 10^{-4}. For BDBEs which exhibit large uncertainties in their frequencies, if the 95%tile of the BDBE frequency exceeds 10^{-4}/plant-year, such BDBEs are evaluated as DBEs. Event sequence families with mean frequencies less than $5x10^{-7}$/plant-year are retained in the PRA results and used to confirm there are no cliff-edge effects and are taken into account in the RI-PB evaluation of defense-in-depth in Task 7e.

Task 5 Select/Revise Safety Related SSCs
Tasks 5 and 6 are performed together rather than sequentially. In Task 6, all the DBEs are subject to a prescriptive evaluation that involves the determination of which safety functions are necessary and sufficient to ensure that 10 CFR 50.34 dose requirements can be met based on a conservative analysis for each safety function challenge represented in each DBE. In Task 5, the design team makes a decision on which SSCs that perform these required safety functions, i.e., those required to be successful to meet the 10 CFR 50.34 dose criteria using conservative assumptions, should be classified as safety related for each DBE. Safety related SSCs are also selected for any required safety function associated with any high consequence BDBEs in which the reliability of the SSC is required to keep the event in the BDBE frequency region. High consequence BDBEs are those with consequences that exceed 10 CFR 50.34 dose criteria. The remaining SSCs that are not classified as safety related are considered in other evaluation tasks including Tasks 7b, 7c, 7d, and 7e. Performance targets and regulatory design criteria for both safety related and non-safety related SSCs are developed and described more fully in a future LMP deliverable on SSC safety classification.

Task 6 Select Deterministic DBAs
For each DBE identified in Task 4, a deterministic DBA is defined that includes the required safety function challenges represented in the DBE, but assumes that the required safety functions are performed exclusively by safety-related SSCs. Non-safety related SSCs are assumed to be failed for each DBA. These DBAs are then used in Chapter 15 of the license application for supporting the conservative deterministic safety analysis. If the design is successful in managing the risks of multiple reactor module accidents, it is expected that DBAs with release of radioactive material will only involve single reactor module accidents and any LBEs involving releases from two or more modules would be BDBEs that would not be risk significant. To achieve this, there should be no DBEs involving a release from two or more modules, and any BDBEs that involve releases from multiple reactor modules would not be high consequence BDBEs. As long as this condition is met the addition of a reactor module to an existing facility should not lead to any new DBAs involving a release of radioactive material.

Task 7 Perform LBE Evaluations
The deterministic and probabilistic safety evaluations that are performed for the full set of LBEs are covered in the following five tasks:

> **Task 7a. Evaluate LBEs against TLRC Frequency – Dose Criteria**
> In this task the results of the PRA which have been organized into LBEs will be evaluated against the TLRC frequency-consequence evaluation criteria of Figure 3-5. The evaluations performed in this task are performed for each LBE separately. The mean values of the frequencies are used to classify the LBEs into AOOs, DBEs, and BDBE categories. However, when the uncertainty bands defined by the 5%tile and 95%tile of the frequency estimates straddles a frequency boundary, the LBE is evaluated in both LBE categories. An LBE with mean frequency above 10^{-}

^{2}/plant-year and 5%tile less than 10^{-2}/plant-year is evaluated as an AOO and DBE. An LBE with mean frequency less than 10^{-4}/plant-year with a 95%tile above 10^{-4}/plant-year is evaluated as a BDBE and a DBE. Uncertainties about the mean values are used to help evaluate the results against the frequency-consequence criteria and to identify the margins against the criteria. This is generally consistent with the LBE approach proposed for NGNP.

Another change in this step relative to that proposed by NGNP in the NGNP LBE White Paper [1] is that DBE doses are evaluated against the frequency-consequence criteria based on the mean rather than the upper 95%tile of the dose uncertainty distribution. This change in approach is based on the fact that the use of conservative dose evaluation is appropriate for the deterministic safety analysis in Task 7a, but is not consistent with the way in which uncertainties are addressed in risk-informed decision making in general. When evaluating risk significance, comparing risks against safety goal QHOs, evaluating changes in risk against RG 1.174 change in risk criteria, the accepted practice has been to first perform a quantitative uncertainty analysis and then to use the mean values to compare against the various goals and criteria.

The primary purpose of comparing the frequencies and consequences of LBEs against the TLRC frequency-consequence curve is to evaluate the risk significance of individual LBEs. The justification for this approach is that uncertainties in the risk assessments have already been reflected in the targets by setting them using conservative assumptions. In summary, the evaluations in this task are based on mean frequencies and mean doses for all three LBE categories. One exception to this is that BDBE's with large uncertainties in their frequencies are evaluated as DBEs when the upper 95%tile of the frequency exceeds 10^{-4} per plant-year; and AOOs with lower 5%tile frequencies below 10^{-4}/plant year are also evaluated as DBEs . The uncertainties about these means are considered as part of the RI-PB evaluation of defense-in-depth in Task 7e.

Part of the LBE frequency-dose evaluation is to ensure that LBEs involving releases from two or more reactor modules do not make a significant contribution to risk and to ensure that measures to manage the risks of multi-module accidents are taken to keep multi-module releases out of the list of DBAs. The need to manage the risks of multi-module accidents is the primary motivation to include such events within the scope of the PRA and in the scope of the SSC performance and design criteria that are influenced by this evaluation.

Finally, another key element of the LBE evaluation in this step is to identify design features that are responsible for keeping the AOOs, and DBEs within their respective frequency-dose criteria including those design features that are responsible for preventing any release for those LBEs with this potential. This evaluation leads to performance requirements and design criteria that are developed within the framework of the SSC classification step in the risk-informed, performance based approach. More discussion of this point is a topic for a future LMP deliverable on SSC safety classification.

Task 7b. Evaluate Integrated Plant Risk against QHOs and 10 CFR 20
In this task the integrated risk of the entire plant including all the LBEs is evaluated against four evaluation criteria including:

- The total frequency of exceeding a site boundary dose of 100 mrem shall not exceed 1/plant-year to ensure that the annual exposure limits in 10 CFR 20 are not exceeded.

- The total frequency of a site boundary dose exceeding 750 rem shall not exceed 10^{-6}/plant-year. Meeting this criterion would conservatively satisfy the NRC Safety Goal Policy Statement [11] on limiting the frequency of a large release.
- The average individual risk of early fatality within 1 mile of the EAB shall not exceed $5x10^{-7}$/plant-year to ensure that the NRC Safety Goal QHO for early fatality risk is met
- The average individual risk of latent cancer fatalities within 10 miles of the EAB shall not exceed $2x10^{-6}$/plant-year to ensure that the NRC safety goal QHO for latent cancer fatality risk is met.

Another key element of this step is to identify design features that are responsible for meeting the integrated risk criteria. This evaluation leads to performance requirements and design criteria that are developed within the framework of the SSC classification step in the TI-RIPB design and licensing approach.

The two QHOs were part of the NGNP LBE approach. This LMP version has added the 10 CFR 20 criterion in recognition that the referenced requirement is for the combined exposures from all releases even though it has been used in developing the TLRC frequency – consequence criteria for evaluating the risks from individual LBEs. Having these cumulative risk criteria as part of the process provides a safeguard to enforce the argument that the TLRC frequency – consequence criteria have been conservatively defined.

Task 7c. Evaluate risk significance of Barriers and SSCs
In this task, the details of the definition and quantification of each of the LBEs in Task 7a and the integrated risk evaluations of Task 7b are used to define both the absolute and relative risk significance of individual SSCs and radionuclide barriers. These evaluations include the use of PRA risk importance metrics, where applicable, and the examination of the effectiveness of each of the barriers in retaining radionuclides. This information is used to provide risk insights to the design team and to support the RI-PB evaluation of defense-in-depth in Task 7e.

Task 7d. Perform Deterministic Safety Analyses against 10 CFR 50.34
This task corresponds to the traditional deterministic safety analysis that is found in Chapter 15 of the license application. It is performed using conservative assumptions. The uncertainty analyses in the mechanistic source terms and radiological doses that are part of the PRA are available to inform the conservative assumptions used in this analysis and to avoid the arbitrary "stacking" of conservative assumptions typically made in traditional deterministic safety analyses.

Task 7e. Risk-Informed, Performance-Based Evaluation of Defense-in-depth
In this task, the definition and evaluation of LBEs will be used to support a RI-PB evaluation of defense-in-depth. This task involves the identification of key sources of uncertainty, and evaluation against defense-in-depth criteria that are the subject of a companion white paper to be developed in the LMP as a future deliverable. Possible outcomes of this task include possible changes to the design, as may be needed, to enhance the plant capabilities for defense-in-depth, formulation of conservative assumptions for the deterministic safety analysis, and input to defining and enhancing programmatic elements of defense-in-depth. It is noted that this DID evaluation does not change the selection of LBEs directly but could lead to the need to change the design or programmatic controls on the design, which in turn would lead to changes in the PRA and thereby affecting the selection of LBEs. Examples of the DID principles to be used in this evaluation have been included in ANS 53.1 and are listed in Table 3-1.

Task 8 Decide on Completion of Design/LBE Development
The purpose of this task is to make a decision as to whether additional design development is needed, either to proceed to the next logical stage of design or to incorporate feedback from the LBE evaluation that design improvements should be considered. Such design improvements could be motivated by a desire to increase margins against the frequency-consequence criteria, reduce uncertainties in the LBE frequencies or consequences, manage the risks of multi-unit accidents, or enhance the performance against defense-in-depth criteria.

Task 9 Proceed to Next Stage of Design Development
The decision to proceed to the next stage of design is reflected in this task. This implies not only completion of the design but also confirmation that defense-in-depth criteria evaluated in Task 7e have been satisfied.

Task 10 Finalize List of LBEs and Safety Related SSCs
Establishing the final list of LBEs and safety related SSCs signifies the completion of the LBE selection process and the selection of the safety related SSCs. The next step in implementing the TI-RIPB approach is to formulate performance requirements and regulatory design criteria for SSCs that are necessary to keep the LBE frequencies and doses within the TLRC frequency-dose criteria. Important information from Task 7b is used for this purpose.

Table 3-1 Defense-in-Depth Principles from ANS 53.1

1	Radionuclide release barriers are sufficiently robust to withstand challenges identified for the design.
2	Each barrier's failure probability is acceptably low compared with identified challenges.
3	As-designed, built, and maintained multiple radionuclide release barriers minimize dependencies. Events that challenge two or more barriers are infrequent, and the postulated failure of one barrier does not significantly increase the failure probability of another barrier.
4	Overall barrier redundancy and diversity ensure compatibility with the TLSC.
5	Accidents potentially releasing significant radioactive material quantities preserve a reasonable prevention/mitigation balance.
6	Safety design avoids overreliance on programs to compensate for plant design weaknesses.
7	System redundancy, independence, and diversity cover expected challenges based on frequency, system failure consequences, and associated uncertainties.
8	The safety design adequately addresses common-cause failures.
9	Performance of a risk-significant safety function is not reliant on a single engineered feature except where inherent safety is demonstrated for all failure modes.
10	The approach evaluates human-error likelihood and consequences, thus providing defenses against human errors that can lead to significant radioactive material release.
11	The design meets the GDC intent applicable in 10 CFR 50, Appendix A, and reactor-specific regulatory design criteria from RI-PB licensing.

3.4.2 Evolution of LBEs through Design and Licensing Stages

The LBE selection flow chart in Figure 3-8 reflects an iterative process involving design development, PRA development, selection of LBEs, and evaluation of LBEs. The process flow chart can be viewed as beginning in the pre-conceptual or conceptual design phase when many design details are unavailable, the PRA effort has not begun, and the safety design approach is just being formulated. To begin the process outlined in Figure 3-8 an initial set of LBEs is proposed based on engineering judgment in Task 1 of the process. This may generate an initial target selection of safety related SSCs.

During the conceptual design phase, different design concepts are explored and alternatives are considered to arrive at a feasible set of alternatives for the plant design. The effort to develop a PRA should begin during this phase. Traditional design and analysis techniques are applied during conceptual design, including (1) use of traditional design bases of engineering analysis and judgment, (2) application of research and development programs, (3) use of past design and operational experience, (4) performance of design trade studies, and (5) decisions on how or whether to conform to established applicable LWR-based reactor design criteria and whether other principle criteria are needed.

Creation of the initial event list of LBEs includes expert evaluation and review of the relevant experience gained from previous reactor designs and associated PRAs, when available. It starts by answering the first question in the risk triplet: What can go wrong? Care must be exercised to ensure that information taken from other reactor technologies is interpreted correctly for the reactor technology in question. The body of relevant reactor design and PRA data that is available to draw upon may vary for different reactor technologies. Once design alternatives and trade studies are developed, the safety design approach can be defined. A review of the major systems can take place and techniques such as a failure modes and effects analysis (FMEAs) and process hazards analyses such as HAZOPs can be applied to identify initial failure scenarios and to support the initial PRA tasks to define initiating events.

Preliminary design activities need to balance regulatory and design requirements, cost, schedule, and other user requirements to optimize the design, cost, and capabilities that satisfy the objectives for the reactor facility.

As the design matures, the scope and level of detail of the PRA is expanded and is used to help support design decisions along the way. An early simplified PRA can be very helpful to support design trade studies that may be performed to better define the safety design approach. Questions that arise in the efforts to build a PRA model may be helpful to the design team especially in the mutual understanding of what kind of challenges will need to be addressed. Because the design is being changed more frequently at this point and better characterized as the design phases evolve, the PRA results and their inputs to the LBE selection process will also be subject to change. As a result, refinements to the list of LBEs are expected. The simplifying perception that a design has stages that contain bright lines is a frequent description at the system level but is not correct at the plant level. Different parts of the design mature at different times. Systems often go through design stages like this, however, at any moment, there may be systems in many design phases simultaneously. Consequently, the PRA development is a continuum as well, maturing with the systems design. PRA updates with system development then provide a more frequent, integrated plant performance check that is otherwise missing in the conventional design process and will also provide risk insights to help the design decisions. When the design, construction, and PRA are developed in a manner that is sufficient to meet PRA requirements reflected in applicable PRA standards and regulatory guides, the LBEs will be finalized and included in the license application.

3.4.3 Role of the PRA in LBE Selection

The primary motivation to utilize inputs from a PRA in the selection of LBEs is that it is the only method available that has the capability to identify the events that are specific and unique to a new reactor design.

Traditional methods for selecting LBEs, such as those reflected in the General Design Criteria and Chapter 15 of the Standard Review Plan, do not refer to a systematic method for identifying design specific events. The generic lists of events provided in the SRP guidance as examples for transients and postulated accidents to consider are specific to LWRs. Traditional systems analysis techniques that can be used to evaluate a design and were used to define the LBEs for currently licensed reactors, including FMEAs, HAZOPs, single failure analyses, etc, have been incorporated into PRA methodology for selecting initiating events and developing event sequence models. PRA is also a mature technology that is supported by industry consensus standards and regulatory guides [31][32][37]. There are no similar consensus standards for deterministic selection of LBEs for new reactor designs. Although much of the available experience in PRA has been with operating LWR plants, there is a rich history of PRA as applied to advanced non-LWR designs including HTGRs, MAGNOX and AGRs, and liquid metal-cooled fast reactors. A trial use PRA standard for advanced non-LWRs was issued by the ASME/ANS Joint Committee on Nuclear Risk Management in 2013 and, by 2018, a revised version for consideration as an ANSI standard is scheduled to be available for ballot. The trial use PRA standard has been subjected to a number of PRA pilot studies on the PRISM, HTR-PM, and several other non-LWR designs. Lessons from these pilot studies are being incorporated into the revised non-LWR PRA standard.

The initial development of the PRA model is closely linked to system engineering analyses that are performed to support the development of the design and the safety design approach. These interfaces are shown in Figure 3-9. It is important to note that the systems engineering inputs on the left hand side of the diagram are fundamental to developing the design. However, with the concurrent development of the PRA model, the PRA is developed in parallel with the design and thereby is available to provide important risk insights to the design development. Decisions to defer the introduction of the PRA to later stages of design lead to reduced opportunities for cost-effective risk management.

The PRA will be used to evaluate the safety characteristics of the design and to provide a structured framework from which the initial set of LBEs will be risk-informed. The evaluation of the risks of the LBEs against the TLRC frequency – consequence criteria help make the LBE selection process both risk-informed and performance-based. This evaluation framework is critical to the development of a revised licensing framework. It highlights the issues that deserve the greatest attention in a safety-focused process. Subsequently the PRA will provide important input to the formulation of performance targets for the capability and reliability of the SSCs to prevent and mitigate accidents and thereby contribute to the performance- based aspects of the design and licensing development process. In addition, engineering judgment and utilization of relevant experience will continue to be used to ensure that LBE selection and classification is complete.

The PRA will systematically enumerate event sequences and assesses the frequency and consequence of each event sequence. Event sequences will include internal events, internal plant hazards, and external events. The modeled event sequences will include the contributions from common cause failures and thereby will not arbitrarily exclude sequences that exceed the single failure criterion.

Each event sequence family reflected in the LBE definitions is defined as a collection of event sequences that similarly challenge plant safety functions. This means that the initiating events within the family have a similar impact on the plant such that the event sequence development following the plant response will be the same for each sequence within the family. If the event sequence involves a release, each sequence in the family will have the same mechanistic source term and offsite radiological consequences. Many of the LBEs do not involve a release, and understanding the plant capabilities to prevent release is an extremely important insight back to the design. Event sequence family grouping facilitates selection of LBEs from many individual events into a manageable number.

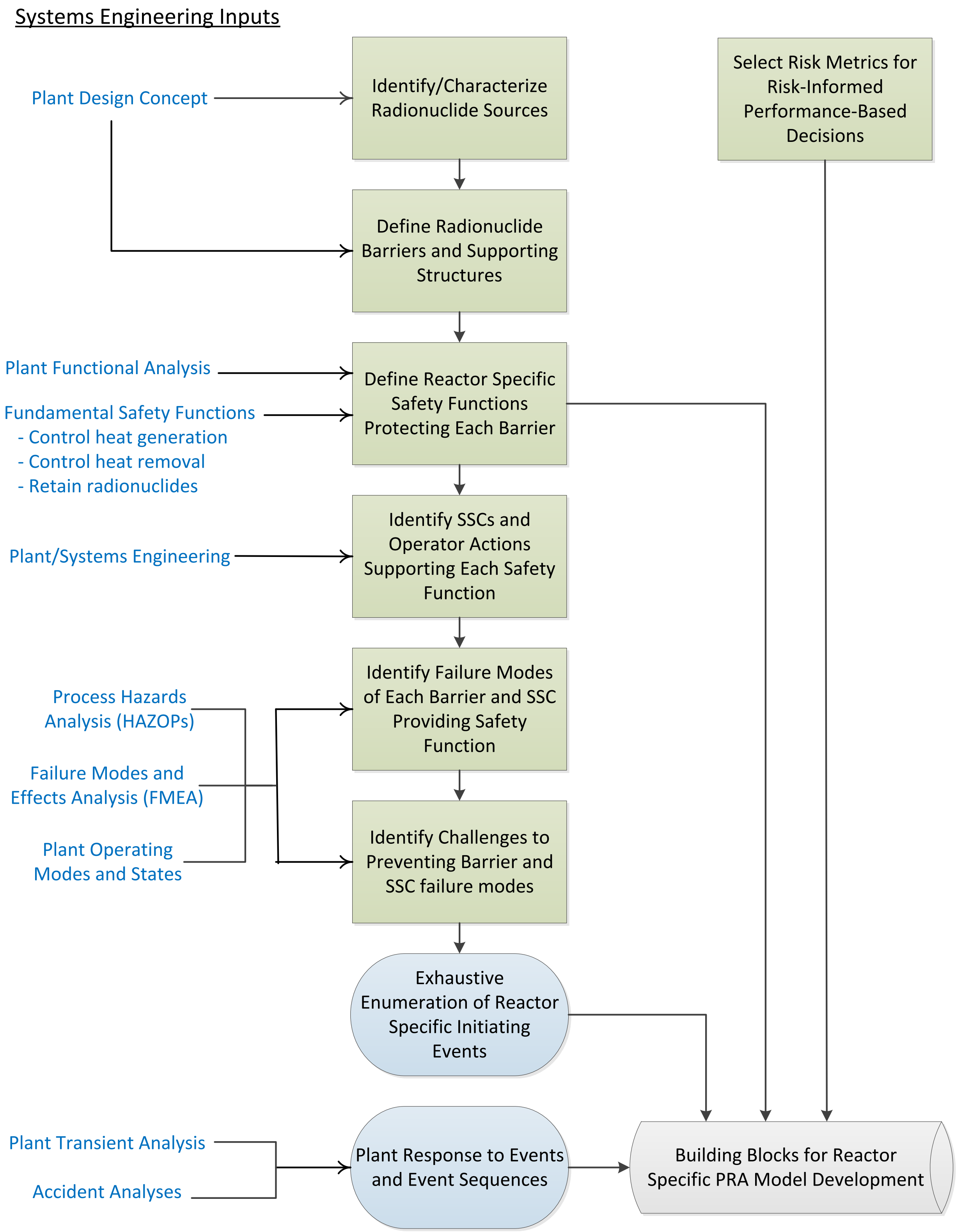

Figure 3-9 Flow Chart for Initial PRA Model Development

The PRA's quantification of both frequencies and consequences will address uncertainties, especially those associated with the potential occurrence of rare events. The quantification of frequencies and consequences of event sequences, and the associated quantification of uncertainties, provides an objective means of comparing the likelihood and consequence of different scenarios against the TLRC. The scope of the PRA, when completed will be as comprehensive and sufficiently complete as a full-scope, all modes, Level 3 PRA covering a full set of internal and external events when the design is completed and site characteristics well defined.

The technical adequacy of the LBE selection process is expected to be enhanced considerably by risk-informing the process. In addition, the PRA will include event sequences involving two or more reactor modules, if applicable, as well as two or more source of radioactive material. This will enable the identification and evaluation of risk management strategies to ensure that sequences involving multiple modules and sources are not risk significant. Because the PRA includes a quantification of offsite radiological consequences, the risk significance of event sequences and SSCs will be made both on more information on the PRA approach to support LBE selection is the topic of a companion paper within the LMP project.

It is recognized that PRA technology has limitations, especially with regard to application to advanced non-LWRs in the design stage. The proposed LBE selection process is not risk-based, but rather risk-informed as there are strong deterministic inputs to the process. First the PRA development is anchored to traditional deterministic system engineering analyses that involve numerous applications of engineering judgment. These are identified in the left side of Figure 3 9 and include FMEAs, process hazards assessment, application of relevant experience from design and licensing of other reactors, and deterministic models of the plant response to events and accidents. Second, the deterministic DBAs are selected based on prescriptive rules and analyzed using conservative assumptions. Finally, the LBE selection includes a review to ensure that the LBE selection and the results of the LBE evaluations meet a set of criteria to ensure the adequacy of defense-in-depth.

These evaluations often lead to changes to the plant design and programmatic controls that are reflected in changes to the PRA and, hence, changes to the selection of LBEs and SSC safety classification. In addition to these elements, peer reviews and regulatory reviews of the PRA will provide an opportunity to challenge the completeness and treatment of uncertainties in the PRA to ensure that the deterministic DBAs and the conservative assumptions that are used in Chapter 15 are sufficient to meet the applicable regulatory requirements. A companion white paper on LMP approach to defense-in-depth and how this approach is integrated into risk-informed and performance-based decision making provides more detail on how deterministic judgments are used to support design and licensing decisions.

3.5 Example Selection of LBEs for HTGRs

In this section, some examples from the MHTGR Preliminary Safety Information Document (PSID) [40] and the supporting PRA [41] are used to illustrate some of the key steps in the LBE definition process of Figure 3-8. The basic steps in LBE definition in this figure were first developed in the MHTGR case. The MHTGR examples presented in this section include several simplified event trees with LBE assignments, examples of AOOs, DBEs, and BDBEs, the process of safety classification of SSCs, and the selection of DBAs. Comparison of the results against the TLRC is also provided. The example LBEs presented in this section for the MHTGR were developed to support a pre-licensing review. The derivation of the LBEs using input from the supporting PRA was documented in a licensing Topical Report [68] which was reviewed by the staff as documented in NUREG-1338.

3.5.1 Example Event Tree Development

The MHTGR PRA included a systematic search for initiating events and included the development and quantification of the frequencies and consequences for the following categories of initiating events:

- Range of HPB failures from small leaks up to offset rupture of relief valve standpipe
- Transients with loss of main loop cooling
- Seismic events
- Loss of offsite power with turbine trip
- Anticipated transients requiring scram
- Inadvertent control rod withdrawal
- Small and large steam generator leaks

Event sequence models were developed based on plant wide thermo-fluid[12] plant response analyses. For event sequences involving a release of radioactive material, mechanistic and event sequence specific source terms and offsite radiological doses were estimated. A full quantification of uncertainties was provided to support the frequency and consequence estimates.

The simplified event tree for very small leaks (< 0.05 in^2) in the MHTGR Helium Pressure Boundary (HPB) is shown in Figure 3-10. The frequencies and probabilities shown in the figure were derived from more detailed event trees in the MHTGR PRA [41]. The MHTGR design is comprised of 4 reactor modules and this initiating event impacts a single module with an estimated frequency of 0.22/plant-year.

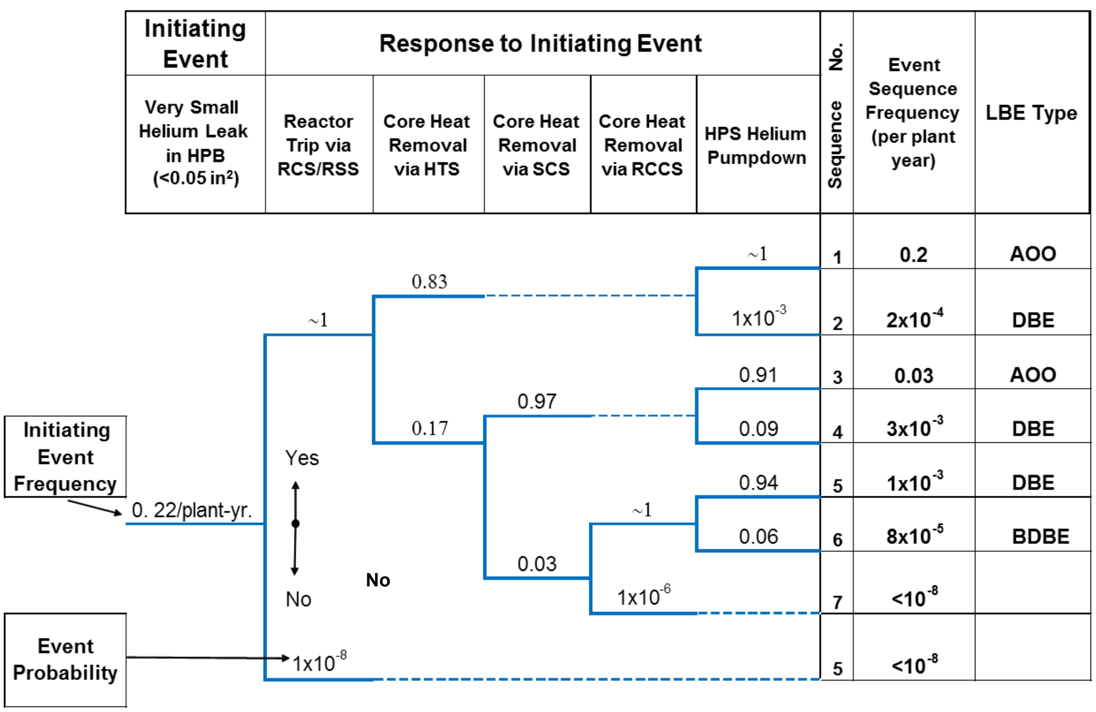

Figure 3-10 Event Tree for MHTGR Very Small Leaks in Helium Pressure Boundary

[12] Because helium is a compressible gas, the term "thermos-fluid" is used in lieu of "thermal hydraulic".

Based on high radiation levels in the reactor building with diverse signals from a reduction in primary system pressure and provisions for manual trip if automatic trip is unsuccessful, there are signals to trip the reactor via the Reactivity Control System (RCS), which is in turn backed up by the diverse Reserve Shutdown System (RSS) yielding a very low probability of failure to insert negative reactivity. Core heat removal is normally provided by continued operation of the main Heat Transport System (HTS). If that method of forced circulation cooling is unavailable, forced circulation cooling is provided by a diverse Shutdown Cooling System (SCS). If both of these systems are unavailable or fail, core heat removal is provided by a passive Reactor Cavity Cooling System (RCCS). Because the leakage of helium is very slow for this initiating event there is sufficient time for the operators to use the Helium Purification and Services System to pump-down the primary system to reduce the leakage from the system and to reduce the pressure drop across the break and the driving force for fluid release to the reactor building.

Based on the frequency of occurrence there are two AOOs involving successful forced cooling and pump-down, two DBEs involving successful forced cooling and failure to effect the pump-down, one DBE with loss of forced cooling with successful passive cooling via the RCCS with successful pump-down, and one BDBE with loss of forced cooling with successful RCCS cooling and no pump-down. Event sequences with frequencies below $5x10^{-7}$/plant-year are not classified as BDBEs but their results are retained in the PRA documentation and evaluated to ensure there are no cliff edge effects. All these LBEs involve a full or partial release of circulating primary coolant radioactivity and those involving loss of forced cooling are also subject to a small delayed fuel release into the reactor building.

An example event tree for an event that challenges all four reactor modules is shown in Figure 3-11. The initiating event for this case is a loss of offsite power and trip of all four turbine generators, each of which is designed to remain on line to supply the house load for AC power. As with the previous example, the event tree includes the expected responses of the RCS and RSS to trip the reactor. Because the initiating event takes out the main HTS possibility for forced cooling, the only option for forced cooling in this case is the SCS. There is one DBE for the case where there is successful forced cooling on all 4 reactor modules and 4 BDBEs in which there is a loss of forced cooling on 1, 2, 3, or all 4 modules.

A third example event tree from the MHTGR PRA is shown in Figure 3-12. The initiating event is an offset rupture of a steam generator tube. The reactor protection systems are designed to detect moisture in the primary system whose signals are backed up by high primary system pressure caused by the moisture ingress to the primary coolant. Additional protection to limit moisture ingress is provided by isolating the secondary side of the SG and then dumping the remaining water and steam into dump tanks inside the reactor building. With successful isolation and continued forced cooling via SCS there is insufficient primary pressure increase to lift the helium pressure relief valves. For failure to isolate or failure of continued forced cooling there is sufficient pressure increase to lift the helium relief valves, which when challenged may open and reclose or may open and fail to close. For sequences in which there is no continued forced cooling and lifting of the helium valves the delayed fuel releases are enhanced somewhat due to the chemical attack on the fuel. This event tree produces one AOO with successful plant response of all functions, one DBE with loss of forced cooling after successful isolation and dump, and three BDBEs only one of which involves a loss of forced cooling.

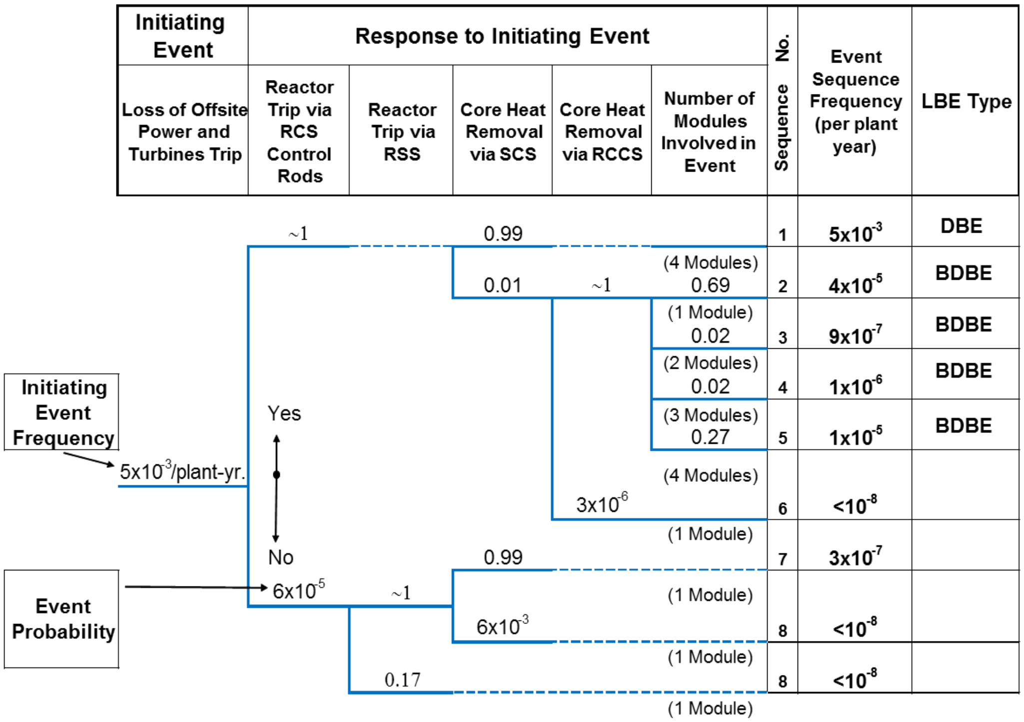

Figure 3-11 Event Tree for MHTGR Loss of Offsite Power and Turbine Trip

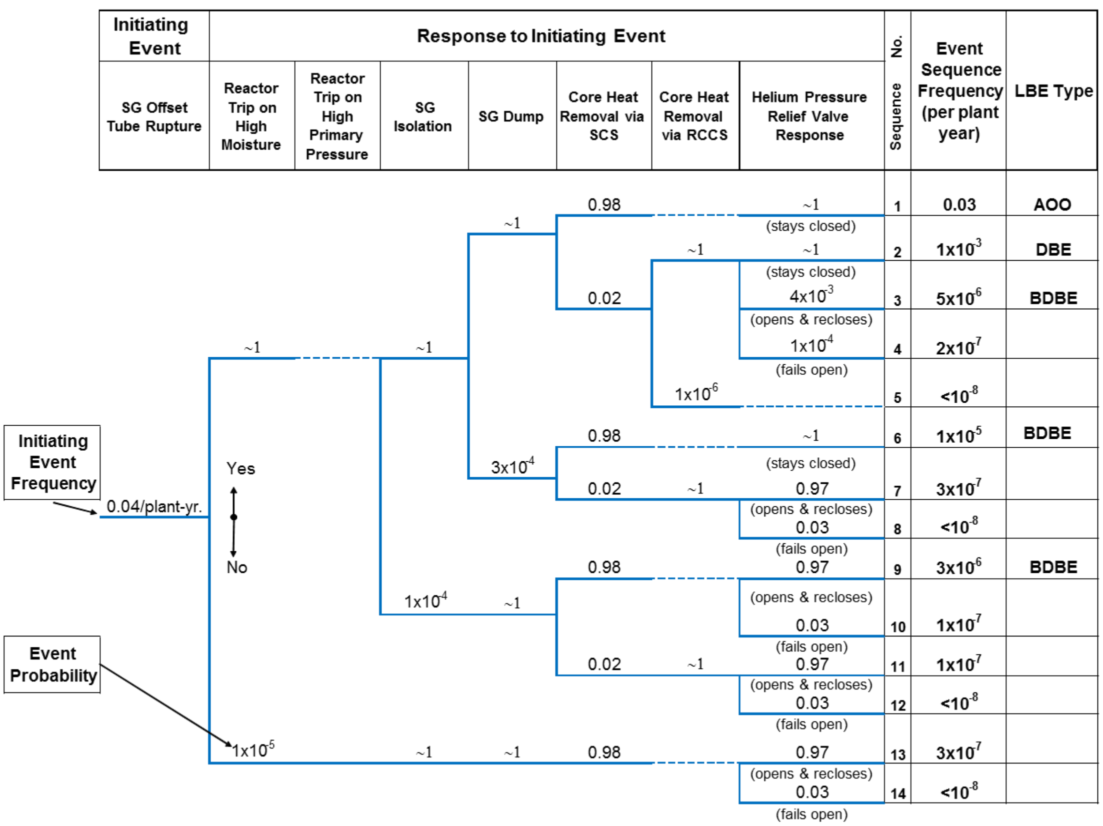

Figure 3-12 Event Tree for MHTGR Steam Generator Tube Rupture

The event trees presented above have been simplified relative to those in the actual PRA in Reference [41] for presentation purposes.

3.5.2 Definition and Evaluation of MHTGR LBEs

Event sequence families are used to group together two or more event sequences when the sequences have a common initiating event, safety function response, and end state. The process of defining event sequence families applies the following considerations:

- The guiding principle is to aggregate event sequences to the maximum extent possible while preserving the functional impacts of the initiating event, safety function responses, and end state. Without event sequence families, excessive detail defining initiating events and balancing event and fault trees could obscure AOO, DBE, or BDBE event sequence classification and yield an unmanageable set of LBEs. By aggregating sequences into family structures, the event sequence model leaves LBE classification essentially unaffected. This approach prevents the problem where very detailed event trees may produce an un-manageable number of LBEs, and the individual event sequence frequencies may be suppressed into the wrong LBE category.
- The safety-function responses are delineated to a necessary and sufficient degree to identify unique challenges to each SSC that performs a given safety function along the event sequence.
- In many cases for a single module plant, there may be only one event sequence in the family.
- For a multi-module plant, event sequence families are used to combine event sequences that involve individual reactor modules independently into a single family of single reactor module event sequences. Event sequences involving multiple reactor modules are always defined as separate LBEs relative to the sequences involving a single module. Accident consequences, where applicable, are evaluated based on the number of reactor modules involved in the release.
- Each event tree initiating event and safety function response has a corresponding fault tree that delineates the event causes and SSC failure modes that contribute to the frequencies and probabilities of these events.
- Many of the LBEs, especially the AOOs and DBEs with relatively high frequencies have zero consequences. Such LBEs are important to identify for the design because they help define the requirements that must be met by SSCs to effect a safe shutdown and prevent a release. This can be contrasted with typical LWR PRAs that focus on sequences that involve core damage and release of radionuclides from the fuel.

After organizing the event sequences in the detailed PRA into accident families having similar initiating events, plant response, and end states, the LBEs in Table 3 2 are defined. A plot of the LBE frequencies, and site boundary doses against the TLRC frequency dose criteria used for the MHTGR are presented in Figure 3 13. Note that the MHTGR used a somewhat different set of TLRC frequency – dose criterion than that proposed in the NGNP LBE White Paper. The key difference was the classification of events as AOOs with frequencies greater than .025/plant-year rather than 10-2/plant-year, the use of annual exposure limits in 10 CFR 50 Appendix I rather than 10 CFR 20, and the use of the LBE designator EPBE for Emergency Planning Basis Events instead of BDBE for beyond design basis events. In addition, both the NGNP and proposed LMP frequency dose criteria are based on the Total Effective Dose Equivalent (TEDE) rather than whole body gamma dose as shown for the MHTGR.

As seen in Figure 3-13, the frequencies and consequences of all the MHTGR LBEs exhibit very large margins against the selected TLRC frequency-consequence criteria. Note that in the MHTGR version of this RI-PB licensing approach, doses were evaluated in terms of whole body gamma doses, whereas in the NGNP and proposed LMP approaches, doses are evaluated in terms of Total Effective Dose Equivalent (TEDE).

Table 3-2 LBEs Identified for the MHTGR [40]

LBE Designation	LBE Description
Anticipated Operational Occurrence	
AOO-1	Transient initiating event with successful reactor trip, continued forced cooling and intact pressurized HPB involving a single reactor module.
AOO-2	Loss of Main Loop Cooling initiating event with successful reactor trip, failure of forced cooling via SCS and intact pressurized HPB involving a single reactor module.
AOO-3	Control Rod Withdrawal with successful control rod trip, continued forced cooling with HTS and intact pressurized HPB involving a single reactor module.
AOO-4	Small SG Leak with successful reactor trip, SG isolation and dump, forced cooling via SCS and intact pressurized HPB involving a single reactor module.
AOO-5	Small HPB Leak with successful reactor trip, continued forced cooling, and successful HPS pump-down, release of part of circulating activity to reactor building involving a single reactor module.
Design Basis Events	
DBE-1	Loss of offsite power initiating event and SCS forced cooling, successful reactor trip, passive cooling via RCCS, intact HPB and no release involving a single reactor module.
DBE-2	Main Loop Transient with Control Rod Trip failure, successful reactor trip via RSS, forced cooling via SCS, intact HPB and no release involving a single reactor module.
DBE-3	Control Rod Withdrawal, with successful reactor trip, Main Loop forced cooling failure, forced cooling via SCS, intact HPB and no release involving a single reactor module.
DBE-4	Control Rod Withdrawal with successful reactor trip, loss of Main and SCS forced cooling via failures, passive cooling via RCCS, intact HPB and no release involving a single reactor module.
DBE-5	Seismic event with loss of offsite power, successful reactor trip, continued forced cooling via Main Loops or SCS, intact HPB and no release involving all four reactor modules.
DBE-6	Moderate SG leak with successful reactor trip, SG isolation and dump, forced cooling via SCS, intact HPB and no release involving a single reactor module.
DBE-7	Moderate SG leak with successful reactor trip, SG isolation and dump, failure of forced cooling via SCS, intact HPB and no release involving a single

LBE Designation	LBE Description
	reactor module.
DBE-8	Moderate SG leak with moisture monitor failure, successful manual reactor trip, SG isolation and dump, forced cooling via SCS, intact HPB and no release involving a single reactor module.
DBE-9	Moderate SG leak with successful reactor trip and SG isolation, failure of SG dump, forced cooling via SCS, circulating activity release via open primary relief valve to reactor building involving a single reactor module.
DBE-10	Moderate HPB leak with successful reactor trip, continued forced cooling, release of circulating activity and lift-off of plateout to reactor building involving a single reactor module.
DBE-11	Small HPB leak with successful reactor trip, failure of forced cooling via Main and SCS Loops, passive cooling via RCCS, partial release of circulating activity and delayed fuel release to reactor building involving a single reactor module.
Beyond Design Basis Events[13]	
BDBE-1 (EPBE-1)	Moderate SG leak with successful reactor trip, delayed SG isolation, SG dump fails, failure of forced cooling via SCS, HPB relief valve opens due to moisture ingress but fails to reseat, HPB depressurizes to reactor building, reactor building vent opens with initial and delayed offsite dose involving a single reactor module.
BDBE-2 (EPBE-2)	Moderate SG leak with successful reactor trip, delayed SG isolation, SG dump fails, successful forced cooling via SCS, HPB relief valve opens due to moisture ingress but fails to reseat, HPB depressurizes to reactor building, reactor building vent opens with initial offsite dose involving a single reactor module.
BDBE-3 (EPBE-3)	Seismic event with small HPB Leaks in all four reactor modules, loss of forced cooling via SCS, release of circulating activity and delayed fuel release to reactor building, HPB depressurizes, reactor building vent opens with offsite dose from release from all four reactor modules.
Notes: HPB = Helium Pressure Boundary SG = Steam Generator SCS = Shutdown Cooling System RCCS = Reactor Cavity Cooling System	

[13] In the MHTGR version of this risk-informed and performance-based licensing approach, BDBEs were referred to as Emergency Planning Basis Events or EPBEs.

3.5.3 Definition of MHTGR DBAs for Chapter 15 Evaluation

DBAs correspond to the traditional off-nromal events analyzed in Chapter 15 of the Safety Analysis Report. The approach in this paper allows the transition to be made from the traditional deterministic plant response with only safety-related SSCs responding to DBAs to all SSCs responding to DBEs, so that both the conservative and expected plant behavior are understood.

As noted in Figure 3-8, to begin the design in Task 1, an initial set of prospective LBEs is identified from which to make some of the initial design decisions. The LBEs are then refined in subsequent tasks based on information provided by the initial PRA.

For consistency with current regulatory requirements, DBAs are identified by assuming that only SSCs classified as safety-related are available to perform the safety functions required to meet 10 CFR §50.34 criteria. The DBAs are defined by examining each of the DBEs and BDBEs and noting which SSCs are available and not available to support each safety function. The designer then selects (Task 5 in Figure 3-8) which SSCs are to be classified as safety related among those available to support each required safety function for each DBE. A required safety function is one that must be fulfilled to meet the 10CFR50.34 dose limits using conservative assumptions. After the safety-related SSCs are selected, all of the DBEs are reanalyzed with only the safety-related SSCs responding in a mechanistically conservative manner. Following this process leads to the definition of DBAs for each of the DBEs in Task 6 in Figure 3-8.

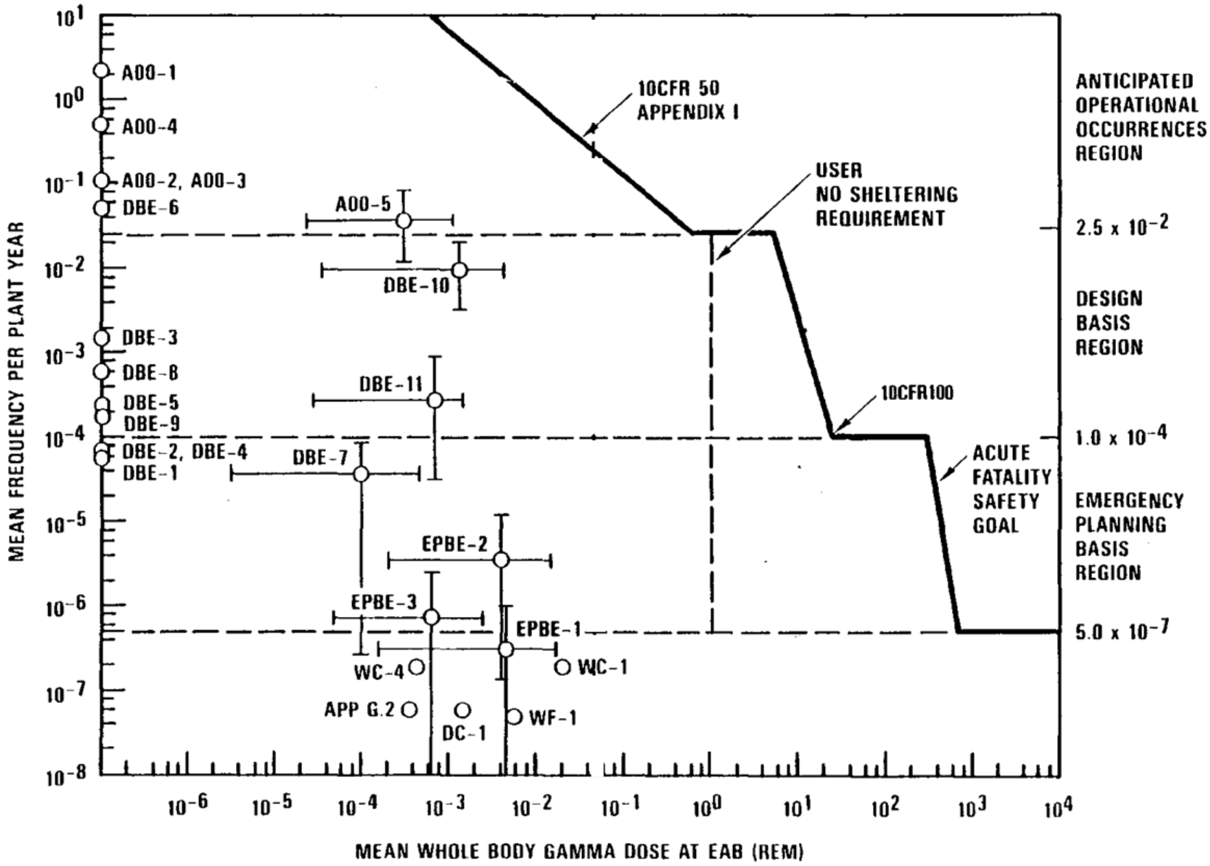

Figure 3-13 Comparison of MHTGR LBE Frequencies and Consequences TLRC Frequency – Dose Criteria[14]

DBAs generally do not have the same sequence of events as corresponding DBEs, since the latter consider the expected plant response with all SSCs responding, whether safety-related or not. This means that some of the DBAs would have frequencies that are lower than the DBE frequency cutoff of 10^{-4}/plant-year.

As noted previously, each DBE is evaluated to identify which SSCs are available and not available support each required safety function, i.e. those safety functions that must be met to maintain the consequences of the DBE within 10 CFR 50.34 dose limits using conservative assumptions. The safety functions defined for the MHTGR, with the required safety functions so designated are shown in Figure 3-14. The development of this figure is based on an exhaustive set of consequence analyses for a wide spectrum of LBEs. One of the required safety functions is core heat removal. To determine which SSCs need to be classified as safety related requires an examination of each of the DBEs and an analysis of which SSCs are available to support that function for each DBE.

[14] EPBE refers to "Emergency Planning Basis Events", the term used in the MHTGR project to denote BDBE

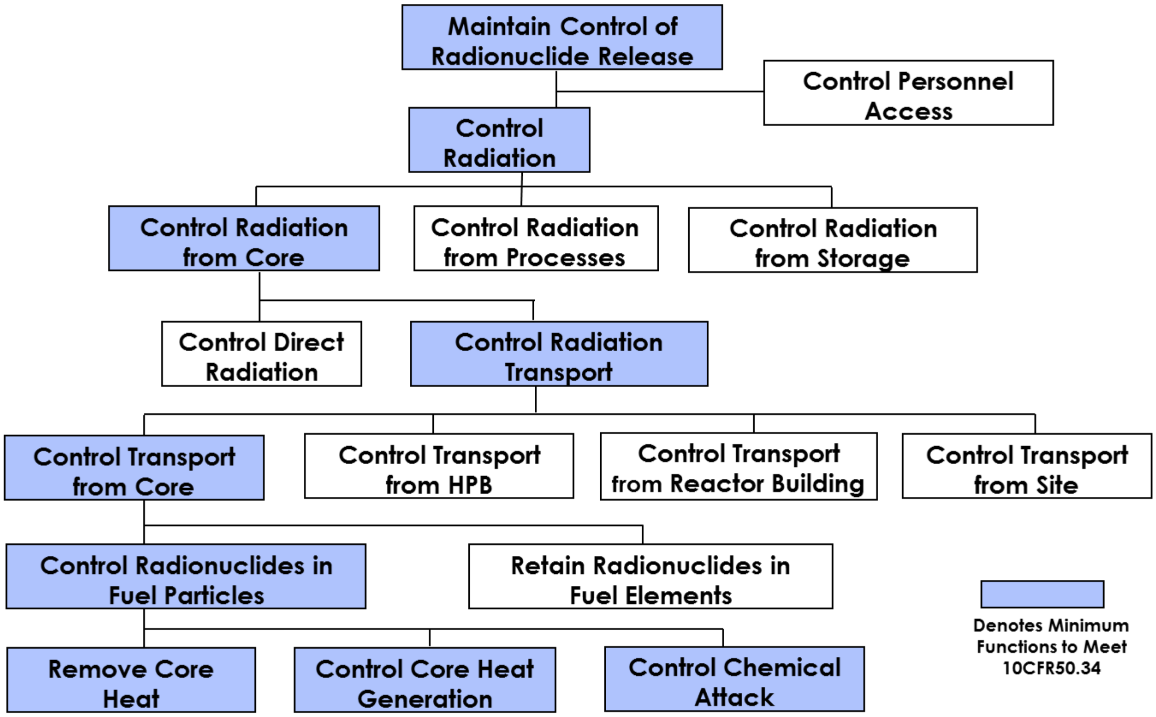

Figure 3-14 MHTGR Safety Functions[15] Including Those Required to Meet 10 CFR 50.34 Limits

Consider DBE-11 that is defined in the small HPB leak event tree in Figure 3-10. The evaluation of the core heat removal SSCs for that DBE is shown in Table 3-3. For this DBE there are two sets of SSCs that are capable of providing this safety function, both involving the reactor and reactor vessel with one transferring heat into the RCCS and the other transferring heat into the passive heat sinks in the reactor cavity of the reactor building.

This evaluation is applied to each of the DBEs to determine which combinations of SSCs are available to support each required safety function. As shown in Table 3-4, there are two options for selecting a set of safety related SSCs that are capable of operation for all of the DBEs. The MHTGR design team selected the combination reactor, reactor vessel, and RCCS as safety-related SSCs. The option that relied on the passive heat sinks in the reactor building as the ultimate heat sink was rejected as that approach involved the need to address uncertainties regarding concrete degradation which are removed with a robust and reliable RCCS. This is an example of how deterministic defense-in-depth considerations had a tangible impact on the selection of safety related SSCs and selection of LBEs.

Table 3-3 Evaluation of Core Heat Removal SSCs for DBE-11

[15] Not shown in this figure is an additional required safety function of "Maintain Core Geometry" which is necessary to for Core Heat Removal and Control of Heat Generation

SSCs Combinations Capable of Providing Core Heat Removal	Available for DBE-11?
• Reactor • Heat Transport System (HTS) • Energy Conversion Area (ECA)	No
• Reactor • Shutdown Cooling System (SCS) • Shutdown Cooling Water System (SCWS)	No
• Reactor • Reactor Vessel (RV) • Reactor Cavity Cooling System (RCCS)	Yes
• Reactor • Reactor Vessel (RV) • Reactor Building passive heat sinks (RB)	Yes

Table 3-4 Evaluation of MHTGR SSCs for Core Heat Removal Safety Function

Alternate Sets of SSCs	Design Basis Events									SSCs Classified as SR?
	DBE 1	DBE 2	DBE 3	DBE 4	DBE 5	DBE 6/7	DBE 8/9	DBE 10	DBE 11	
• Reactor • HTS • ECA	No	No	No	No	No	No	No	No	No	No
• Reactor • SCS • SCWS	No	Yes	Yes	No	Yes	Yes	Yes	Yes	No	No
• Reactor • RV • RCCS	**Yes**	**Yes**	**Yes**	**Yes**	**Yes**	**Yes**	**Yes**	**Yes**	**Yes**	**Yes**
• Reactor • RV • RB	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	No

When this process is completed for each required safety function, it is possible to define DBAs for eachof the DBEs where only safety related SSCs are assumed to be operable and all the non-safety related SSCs are assumed to be failed. The DBAs defined for the MHTGR are shown in Table 3-5.

In the course of defining these DBAs, the MHTGR design team classified the following additional SSCs as safety related for the other shaded functions in Figure 3-14: the RCS and RSS reactor trip systems, the moisture monitors and SSCs necessary to ensure successful isolation of a leaking steam generator, but not the SG dump system. There are three DBEs for which the corresponding DBA is the same because the required safety functions are performed by the safety related SSCs for that function. For DBE-3 and DBE-4, the same DBA is defined. Also for DBE-6, DBE-7, DBE-8 and DBE-9 the same DBA is defined. So there are fewer different DBAs than DBEs. This stems from the fact that some of the DBEs have successful operation of one or more non-safety SSCs to perform a required safety function, whereas DBAs only have safety related SSCs assumed to be operational for such functions.

Each of the DBAs are then included in Chapter 15 of the license application and are analyzed using conservative assumptions and demonstrated to meet 10 CFR 50.34 dose limits.

Table 3-5 Definition of Deterministic DBAs for MHTGR

<table>
<tr><th>DBE</th><th>Design Basis Events</th><th>DBA</th><th>Design Basis Accidents</th></tr>
<tr><td>DBE-1</td><td>Loss of offsite power initiating event and SCS forced cooling, successful reactor trip, passive cooling via RCCS, intact HPB and no release involving a single reactor module. (corresponds to PRA sequence family with frequency of $5x10^{-5}$/plant-year or about $1x10^{-5}$/reactor-year)</td><td>DBA-1</td><td>Loss of Main and SCS forced cooling, successful reactor trip, passive cooling via RCCS, intact HPB and no release involving a single reactor module (corresponds to PRA sequence family with frequency of $5x10^{-5}$/plant-year or about $1x10^{-5}$/reactor-year)</td></tr>
<tr><td>DBE-2</td><td>Main Loop Transient with Control Rod Trip failure, successful reactor trip via RSS, forced cooling via SCS, intact HPB and no release involving a single reactor module. (corresponds to PRA sequence family with frequency of $7x10^{-5}$/plant-year or about $2x10^{-5}$/reactor-year)</td><td>DBA-2</td><td>Loss of Main and SCS forced cooling with Control Rod Trip failure, successful reactor trip via RSS, passive cooling, intact HPB and no release involving a single reactor module. (corresponds to PRA sequence family with frequency of $7x10^{-5}$/plant-year or about $2x10^{-5}$/reactor-year)</td></tr>
<tr><td>DBE-3</td><td>Control Rod Withdrawal, with successful reactor trip, Main Loop forced cooling failure, forced cooling via SCS, intact HPB and no release involving a single reactor module. (corresponds to PRA sequence family with frequency of $2x10^{-3}$/plant-year or about $5x10^{-4}$/reactor-year)</td><td rowspan="2">DBA-3
DBA-4</td><td rowspan="2">Control Rod Withdrawal, with successful reactor trip, failure of forced cooling via Main loops and SCS, passive cooling via RCCS, intact HPB and no release involving a single reactor module. (corresponds to PRA sequence family with frequency of $7x10^{-5}$/plant-year or about $2x10^{-5}$/reactor-year)</td></tr>
<tr><td>DBE-4</td><td>Control Rod Withdrawal with successful reactor trip, loss of Main and SCS forced cooling via failures, passive cooling via RCCS, intact HPB and no release involving a **single** reactor module. (corresponds to PRA sequence family with frequency of $7x10^{-5}$/plant-year or about $2x10^{-5}$/reactor-year)</td></tr>
</table>

DBE	Design Basis Events	DBA	Design Basis Accidents
DBE-5	Seismic event with loss of offsite power, successful reactor trip, continued forced cooling via Main ~~Loop~~s or SCS, intact HPB and no release involving all four reactor modules. (corresponds to PRA sequence family with frequency of $2x10^{-4}$/plant-year or $2x10^{-4}$/reactor-year)	DBA-5	Seismic event with loss of offsite power, successful reactor trip, failure of forced cooling via Main ~~Loops~~ ~~or~~ and SCS, passive cooling via RCCS, intact HPB and no release involving all four reactor modules. (corresponds to PRA sequence family with frequency of $6x10^{-8}$/plant-year or ~$6x10^{-8}$/reactor-year)
DBE-6	Moderate SG leak with successful reactor trip, SG isolation and dump, forced cooling via SCS, intact HPB and no release involving a single reactor module. (corresponds to PRA sequence family with frequency of $5x10^{-2}$/plant-year or about $1x10^{-2}$/reactor-year)	DBA-6	Moderate SG leak with successful reactor trip and SG isolation, failure of SG dump, failure of forced cooling via SCS, passive cooling via RCCS, circulating activity and delayed fuel release via primary relief valve to reactor building involving a single reactor module. (corresponds to PRA sequence family with frequency of 2x10-7/plant-year or $5x10^{-8}$/reactor-year)

DBE	Design Basis Events	DBA	Design Basis Accidents
DBE-7	Moderate SG leak with successful reactor trip, SG isolation and dump, failure of forced cooling via SCS, intact HPB and no release involving a single reactor module. (corresponds to PRA sequence family with frequency of $4x10^{-5}$/plant-year or $1x10^{-5}$/reactor-year)	DBA-7 DBA-8 DBA-9	Moderate SG leak with successful reactor trip and SG isolation, failure of SG dump, failure of forced cooling via SCS, passive cooling via RCCS, circulating activity and delayed fuel release via primary relief valve to reactor building involving a single reactor module. (corresponds to PRA sequence family with frequency of $<10^{-8}$/plant-year or $<10^{-8}$/reactor-year)
DBE-8	Moderate SG leak with moisture monitor failure, successful manual reactor trip, SG isolation and dump, forced cooling via SCS, intact HPB and no release involving a single reactor module. (corresponds to PRA sequence family with frequency of $4x10^{-5}$/plant-year)		
DBE-9	Moderate SG leak with successful reactor trip and SG isolation, failure of SG dump, forced cooling via SCS, circulating activity release via open primary relief valve to reactor building involving a single reactor module. (corresponds to PRA sequence family with frequency of $2x10^{-4}$/plant-year)		
DBE-10	Moderate HPB leak with successful reactor trip, continued forced cooling, release of circulating activity and lift-off of plateout to reactor building involving a single reactor module. (corresponds to PRA sequence family with frequency of $1x10^{-2}$/plant-year or about $3x10^{-3}$/reactor-year)	DBA-10	Moderate HPB leak with successful reactor trip, failure of forced cooling via Main loops and SCS, passive cooling via RCCS, release of circulating activity, delayed fuel release, and lift-off of plateout to reactor building involving a single reactor module. (corresponds to PRA sequence family with frequency of $6x10^{-8}$/plant-year or about $1.5x10^{-8}$/reactor-year)
DBE-11	Small HPB leak with successful reactor trip, failure of forced cooling via Main and SCS Loops~~,~~ passive cooling via RCCS, partial release of circulating activity and delayed fuel release to reactor building involving a single reactor module. (corresponds to PRA sequence family	DBA-11	Small HPB leak with successful reactor trip, failure of forced cooling via Main and SCS, partial release of circulating activity and delayed fuel release to reactor building involving a single reactor-module. (corresponds to PRA sequence family with frequency of

DBE	Design Basis Events	DBA	Design Basis Accidents
	with frequency of $3x10^{-4}$/plant-year or about $8x10^{-5}$/reactor-year)		$<10^{-8}$/plant-year or $<10^{-8}$/reactor-year)

3.6 Example LBE Development for PRISM

In this section, some examples from the PRISM PRA [69] used to illustrate some of the key steps in the LBE definition process of Figure 3-8. The PRISM example presented in this section is a simplified event tree for a loss of forced flow event. Examples of AOOs, DBEs, and BDBEs are also taken from the PRISM PRA and the process of selecting options for safety-related SSCs is demonstrated. It is noted that PRISM has not had the benefit of the full application of the LBE selection and SSC safety classification process as presented in the previous section for the MHTGR. These PRISM examples benefit from a recently completed PRA upgrade that was performed for PRISM, which served as one of the pilot PRAs for the Advanced non-LWR Trial Use PRA Standard [32]. More examples from the PRISM PRA are included in a companion LMP paper on PRA development for advanced non-LWRs.

3.6.1 Example Event Tree Development

The PRISM plant is comprised of two reactor modules whose reactors are of the pool-type liquid metal cooled fast reactor similar in design to EBR-2.

The PRISM PRA included a systematic search for initiating events and included the development and quantification of the frequencies and consequences for the following categories of initiating events:

- BOP/Loss of Heat Sink (LOHS) faults
- Intermediate Heat Exchanger (IHX) bypass leak
- Intermediate Heat Transport System (IHTS) leak
- Loss of Offsite Power (LOOP)
- Loss of Primary Forced Flow (LOF)
- NSSS transients
- Turbine/BOP transient faults
- Steam Generator Tube Rupture (SGTR)
- Transient overpower

Event sequence models were developed based on the challenges that the initiating events placed on radionuclide release barriers. Plant response analysis includes assessments of structural integrity, thermal-hydraulic sodium system temperatures, and fuel performance. For sequences involving any damage to fuel cladding to the core assemblies or spent fuel stored in the vessel, a mechanistic source term analysis calculates the radionuclide transport from the fuel, to the sodium coolant hot pool, to the cover gas space, through leakage paths in the vessel, and through leakage paths in containment into the environment.

The simplified event tree for LOF from a single electromagnetic (EM) pump failure is provided in Figure 3-15. With conservative estimates for various failure modes across the eight EM pumps across the two reactor units, the total frequency of LOF from a single pump is about once per plant-year.

The loss of flow is immediately detected by redundant pressure sensors at the discharge points for each of the EM pumps, which triggers a scram in the Reactor Protection System (RPS). Because of redundancy in the RPS architecture, the dominant scram failure mode is a common cause software failure of digital instrumentation and controls, assessed as 1E-4 per demand. Failure of RPS is accommodated by a Diverse Protection System (DPS), which is a digital system on a completely independent platform, also dominated by a common cause software demand failure. Overall, the failure of scram following the LOF initiating event has an event sequence frequency below the range of Beyond Design Basis Events (BDBE).

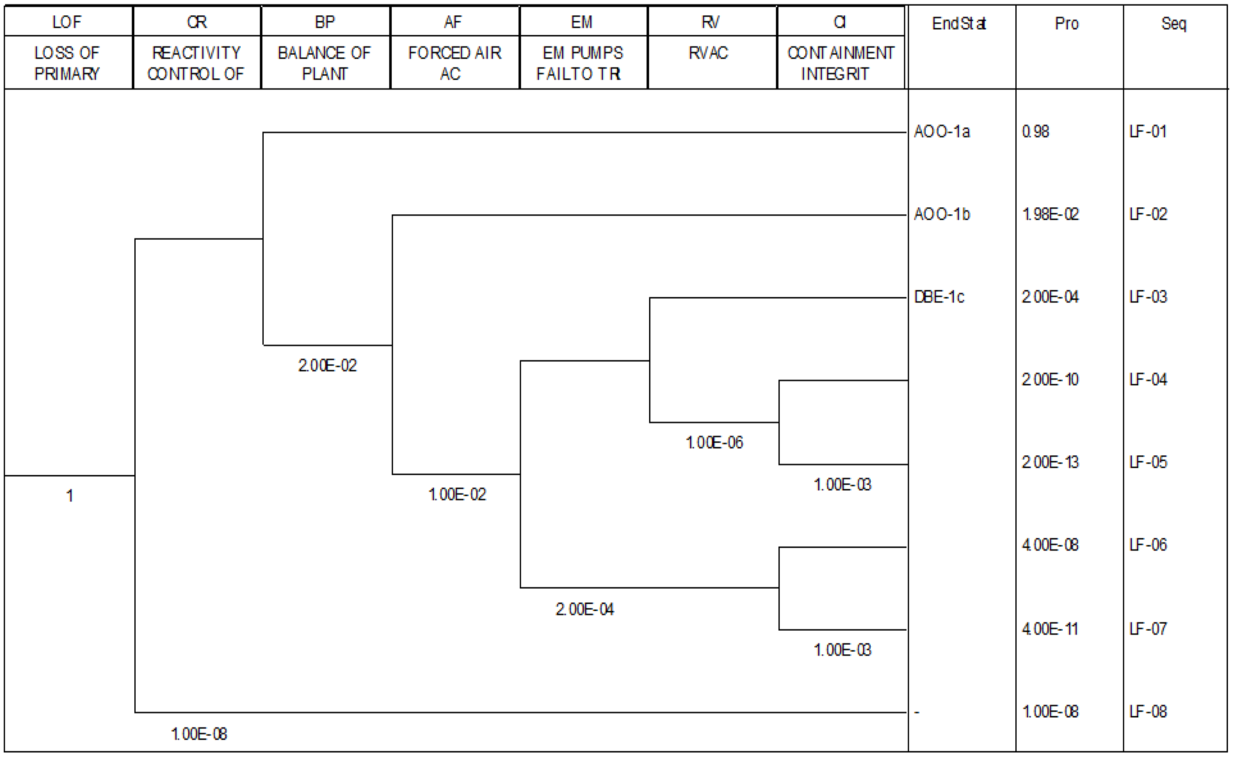

Figure 3-15 Event Tree for Loss of Flow in a Single EM Pump

Once the reactivity control function is performed by successful control rod scram, the safety function of interest is decay heat removal. The first system satisfying this function is the normal compliment of feedwater and condensate systems to remove decay heat through the steam generator. This event represents AOO-1a. In the event that these BOP systems fail, decay heat could be removed by manual actuation of a fan that circulates air around the shell of the steam generator. This forced air cooling mode of the Steam Generator Auxiliary Cooling System (SGACS) can remove all decay heat transferred by the intermediate sodium cooling loop to the steam generator. This is event AOO-1b.

On failure of both the BOP and SGACS active systems, decay heat removal can be accomplished by the Reactor Vessel Auxiliary Cooling System (RVACS), which consists of pathways for air to circulate around the outside of the containment vessel. RVACS, which is a set of plant structures rather than a system, is capable of removing decay heat loads entirely with passive natural air circulation. In order fulfill this function, the EM pumps that drive primary and intermediate coolant need to have successfully tripped. Each of the pumps are self-cooled, meaning that their full work load is transferred to the sodium coolant. This event sequence where a LOF initiating event is followed by failures of BOP cooling and the backup SGACS fan is DBE-1c.

The LOF event tree contains several events where fuel damage and thus release is possible. The event LF-04 is the result when RVACS passive flow conditions are degraded in such a manner that temperatures in the vessel climb and challenge fuel integrity criteria. Passive reliability analysis for RVACS justifies that many off-normal structural conditions are needed to degrade RVACS performance so severely, resulting in an overall reliability on the order of 1E-6 per demand. After fuel damage, a very small radionuclide release in this event occurs as the result of design leakage out of the vessel and then out of the containment. The frequency of this event sequence has been judged to be below the range of BDBEs.

The LOF event tree had yet another unlikely sequence postulating fuel damage. This is the event (LF-06) where BOP cooling and SGACS fail, and there is a failure of a minimum number of EM pumps to trip. If the EM pumps are allowed to continue running, the heat load on the sodium coolant is larger than the capacity of RVACS heat removal, resulting in eventual fuel damage. Similar to LF-04 discussed above, this event results in a small release through design leakage paths, as calculated by a mechanistic source term analysis. This event is also well below the range of BDBEs.

The collection of event sequences following the LOF initiator is typical for the decay heat removal function following any initiating event in the PRISM plant.

3.6.2 Definition and Evaluation of PRISM LBEs

The AOOs and DBE identified in the LOF event tree above are collected in Table 3-6, alongside some examples of BDBEs identified from other event trees not discussed here. A plot of the LBE frequencies, and site boundary doses against the TLRC frequency dose criteria used for PRISM are presented in Figure 3-16.

Similar to the conclusion for MHGTR, the frequencies and consequences of all the PRISM LBEs exhibit very large margins against the selected TLRC frequency-consequence criteria. This conclusion is expected to remain true when the full list of events studied in the PRA are added.

Table 3-6 LBEs Identified for the PRISM Loss of Flow Event Tree

LBE Designation	LBE Description
Anticipated Operational Occurrence	
AOO-1a	Transient initiating event with successful reactor trip and successful cooling through Balance of Plant (BOP) systems; no fuel damage
AOO-1b	Transient initiating event with successful reactor trip, failure of BOP cooling, but success of forced-air Steam Generator Auxiliary Cooling System; no fuel damage
Design Basis Events	
DBE-1c	Transient initiating event with failure of active decay heat removal, but success of passive air-cooling with RVACS; no fuel damage
Beyond Design Basis Events	
BDBE-2	Spurious control rod withdrawal with successful scram, failures of decay heat removal through both BOP systems and SGACS, but successful passive air-cooling with RVACS; no fuel damage
BDBE-3	Steam generator tube rupture event with successful scram and suppression of sodium water reaction, but failure of SGACS; RVACS is successful in this event at removing decay heat and there is no fuel damage

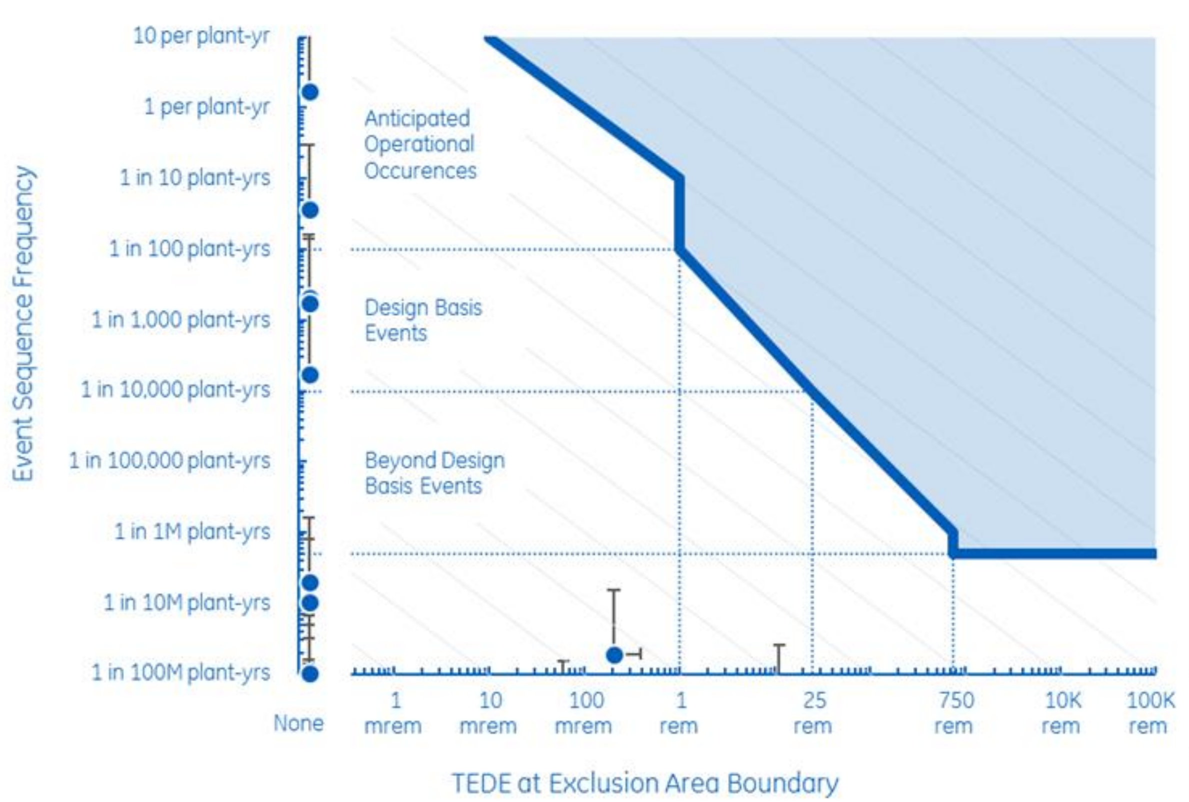

Figure 3-16 Comparison of PRISM LBE Frequencies and Consequences and TLRC Frequency – Dose Criteria

3.6.3 Example Definition of PRISM DBAs

With the events taken from the Internal Events At-Power PRA for PRISM, there are a limited number of events that fall in the DBE range. Of these events, nearly all fit the pattern of some sort of plant trip, followed by failures of multiple active cooling systems, leaving the decay heat removal function to the passive air-cooling with RVACS, as in the DBE example discussed above.

For DBE-1c, which is a typical decay heat removal challenge, the combinations of SSCs needed to meet dose release limits set by 10 CFR 50.34 are provided in

Table 3-7. The table shows that there are three primary ways to remove decay heat and avoid fuel damage: BOP cooling, forced-air cooling through SGACS, or passive air cooling through RVACS. A fourth option is provided for completeness, leveraging the mechanistic source term analysis that shows that given fuel damage from lack of cooling, the dose release could meet 10 CFR 50.34 limits if both the vessel head and containment perform their confinement function successfully.

It should be noted that these PRISM example DBAs were developed by taking a pre-existing PRA that was originally focused on the identification of event sequences with releases and consequences. Had the PRA been originally developed for the purpose of identifying LBEs the challenges to the safety functions that do not involve a release would have been more fully developed resulting in additional DBAs.

Table 3-7 Evaluation of SSCs Limiting Dose Release for PRISM DBE-1c

SSCs Combinations Capable of Meeting 10 CFR 50.34 Dose Limits	Available for DBE-1c?
• Reactor vessel • Control rod scram • BOP cooling • RVACS (passive air-cooling)	No
• Reactor vessel • Control rod scram • SGACS cooling	No
• Reactor vessel • Control rod scram • RVACS passive air-cooling	Yes
• Reactor vessel • Control rod scram • No decay heat removal (fuel damage) • Vessel head • Containment	Yes

A more complete set of DBEs and corresponding DBAs for PRISM is shown in Table 3-8.

Table 3-8 Definition of Deterministic DBAs for PRISM

DBE	Design Basis Events	DBA	Design Basis Accidents
DBE-01	Spurious control rod withdrawal with pre-existing rod-stop error and failure of BOP cooling; RPS shuts down reactor and active SGACS removes decay heat involving one reactor module	DBA-01	Spurious control rod withdrawal with pre-existing rod-stop error and failure of BOP cooling and forced SGACS cooling; RPS shuts down the reactor and passive RVACS removes decay heat, including the extra power generated during the transient overpower involving one reactor module
DBE-02	Spurious control rod withdrawal with pre-existing rod-stop error and failure of BOP cooling and SGACS; RPS shuts down reactor, the EM pumps trip, and passive RVACS removes decay heat, supplemented by passive mode of SGACS involving one reactor module		
DBE-03	Spurious control rod withdrawal with pre-existing rod-stop error and failure of BOP cooling and SGACS; RPS shuts down reactor, the EM pumps trip, and passive RVACS removes decay heat involving one reactor module		
DBE-04	Steam generator tube rupture is detected and suppressed by sodium-water reaction detection equipment, RPS shuts down the reactor, and active SGACS removes decay heat involving one reactor module	DBA-02	Steam generator tube rupture with failure of sodium-water reaction detection and suppression equipment, which disables all cooling modes through the intermediate loop; RPS shuts down the reactor and passive RVACS removes decay heat involving one reactor module
DBE-05	A general transient with failure of BOP cooling and forced SGACS; RPS shuts down the reactor, the EM pumps trip, and passive RVACS removes decay heat supplemented by passive mode of SGACS involving both reactor modules	DBA-03	A general transient with failure of BOP cooling and forced SGACS; RPS shuts down the reactor, the EM pumps trip and passive RVACS removes decay heat involving both reactor modules
DBE-06	A general transient with failure of BOP cooling and all modes of SGACS; RPS shuts down the reactor, the EM pumps trip, and passive RVACS removes decay		

DBE	Design Basis Events	DBA	Design Basis Accidents
	heat involving both reactor modules		
DBE-07	A general transient with failure of the intermediate sodium coolant loop; RPS shuts down the reactor, the EM pumps trip, and passive RVACS removes decay heat involving both reactor modules		
DBE-08	A plant-centered loss of offsite power with failure of backup power to forced SGACS; RPS shuts down the reactor and passive RVACS removes decay heat involving both reactor modules		
DBE-09	A major hurricane causes both a loss of offsite power and an off-normal condition for RVACS; RPS shuts down the reactor and passive RVACS removes decay heat under storm conditions involving both reactor modules	DBA-04	A major hurricane causes both a loss of offsite power and an off-normal condition for RVACS; RPS shuts down the reactor and passive RVACS removes decay heat under storm conditions involving both reactor modules

3.7 LMP LBE Selection Approach Summary

In summary, the regulatory precedents were reviewed to identify guidance for selecting LBEs for advanced non-LWRs. Example approaches for selecting LBEs were identified from the Department of Energy NGNP project and NUREG-1860. Similar approaches were identified with the Yucca Mountain Preclosure Safety Analysis and the United Kingdom SAPs. The regulatory precedent review benefitted from more recent developments since the NGNP LBE White Paper was developed and provided additional insights into the formulation of frequency-consequence criteria for evaluating the risk significance of selected LBEs.

Based on the review of the regulatory precedents and consistent with the LMP objectives, a set of LBE selection attributes were developed. To meet these attributes, the LBE selection approach shall be:

- **Systematic and Reproducible**.
- **Reasonably Complete**
- **Provide Timely Input to Design Decisions**
- **Risk-informed and Performance Based**
- **Reactor Technology Inclusive**
- **Consistent with Applicable Regulatory Requirements**

An approach to selecting LBEs for the LMP was selected using the NGNP LBE as a starting point. An earlier version of the NGNP approach was successfully applied in the MHTGR project which included the development of a conceptual design of an MHTGR plant consisting of four reactor modules, a multi-module PRA, a Preliminary Safety Information Document, and several topical reports that described the derivation of LBEs and the selection of safety related SSCs. This information was subjected to a preliminary review by the NRC staff and supporting national laboratories. The risk-informed and performance based approach reflected in this MHTGR case was subsequently refined in the Exelon PBMR, and NGNP projects.

With the benefits of an expanded regulatory precedent review and the feedback from the NRC and ACRS on the NGNP and supporting projects, a number of refinements are proposed in the LMP LBE selection approach. These refinements have benefited from developing examples LBE selections for the PRISM, a liquid metal cooled, pooled type fast reactor to ensure that the selected approach is reactor technology inclusive. The refinements include:

- Refinement to the TLRC frequency-dose criteria for evaluating the risk significance of individual LBEs
- Addition of two risk metrics and associated performance goals to evaluate the integrated risks of the multi-module advanced non-LWR plant, beyond the two QHO risk metrics used in the NGNP approach. The two additional metrics address a goal for managing LRF below 10^{-6}/plant-year and a goal for ensuring that the annual dose limits in 10 CFR 20 are met.

Beyond these changes, the LBE approach that is proposed is consistent with that originally developed for the MHTGR and subsequently refined during the Exelon PBMR and NGNP projects. It is the view of the LMP project team that the LBE approach described in this Section has the LBE selection attributes listed above and has the capability to derive a set of LBEs that will be necessary and sufficient for the design and licensing of advanced non-LWR plants.

The proposed approach is systematic and reproducible. It has been demonstrated to be reactor technology inclusive using examples from two distinctly different reactor types, a modular MHTGR and a pool-type liquid metal cooled fast reactor, PRISM. It is risk informed because it employs an appropriate balance of deterministic and probabilistic inputs and is consistent with the principles of defense-in-depth. The approach is performance based at the plant level through the use of TLRC frequency-consequence criteria that can be calculated and compared against the risk targets. This process leads to RIPB practices at lower tier activities including design specific principle design criteria development, RIPB SSC classification and capability, and a well-structured framework for DID evaluation. While addressing the fundamental differences between advanced non-LWRs and LWRs the approach is designed to address applicable regulatory requirements and has the capability to support successful future license applications that reflect the NRC Safety Goal and Advanced Reactor Policies.

4. REVIEW OF OUTCOME OBJECTIVES

The information provided in this white paper is intended to serve as the basis for interaction with the NRC staff. Section 1.4 introduced a set of outcome objectives that require interactions with the NRC regarding selection and classification of LBEs.

The LMP Project is seeking:

(1) NRC's approval of the proposed LBE selection approach for incorporation into appropriate regulatory guidance;
(2) Identification of any issues that have the potential to significantly impact the selection and evaluation of LBEs, including anticipated operational occurrences (AOOs), design basis events (DBEs), beyond design basis events (BDBEs) and design basis accidents (DBAs)

The following are specific areas where agreement on the LMP Project's approach to the selection and classification of LBEs is being sought.

A summary of the LMP approach for each outcome objective, which is described in detail in Section 3 of this report, is also provided.

The LMP is seeking agreement for the following specific areas:

- The structured, TI-RIPB process described in this document is an acceptable approach for defining the LBEs for advanced non-LWRs such as modular HTGRs, molten salt reactors, and liquid metal cooled reactors. A means of documenting NRC review and approval of this approach is an essential outcome objective.

  <u>LMP Approach:</u>

  The LMP approach is based on the LBE approach developed for the MHTGR and subsequently refined during the Exelon PBMR and NGNP projects. Several refinements to the LMP approach are proposed in this paper in order to address lessons learned from NRC and ACRS staff reviews of the LMP method and to ensure that the key attributes of the LBE selection process are addressed. These attributes call for an approach that is TI-RIPB, reproducible, reasonably complete, consistent across technologies, and capable of identifying the appropriate LBEs for each reactor design and technology.
- The LMP approach to defining LBEs described herein is appropriate. As the term is used in this document, LBEs are defined broadly to include all the events used to support the design and meet licensing requirements. They cover a comprehensive spectrum of events from normal operation to rare, off-normal events.

  <u>LMP Approach</u>

  There are four categories of LBEs:

  - Anticipated Operational Occurrences (AOOs), which encompass planned and anticipated events. The radiological doses from AOOs are required to meet normal operation public dose requirements. AOOs are utilized to set evaluation criteria for normal operation modes and states.
  - Design Basis Events (DBEs) encompass unplanned off-normal events not expected in the plant's lifetime, but which might occur in the lifetimes of a fleet of plants. The radiological doses from DBEs are required to meet accident public dose requirements. DBEs are the basis for the design, construction, and operation of the structures, systems, and components (SSCs) during accidents.

- Beyond Design Basis Events (BDBEs), which are rare off-normal events of lower frequency than DBEs. BDBEs are evaluated to ensure that they do not pose an unacceptable risk to the public.
- Design Basis Accidents (DBAs). The DBAs for Chapter 15, "Accident Analyses," of the license application are deterministically derived from the DBEs by assuming that only SSCs classified as safety-related are available to mitigate the consequences. The public consequences of DBAs are based on mechanistic source terms and are conservatively calculated. The upper 95% conservative estimate of the dose of each DBA must meet the 10 CFR §50.34 consequence limit at the Exclusion Area Boundary (EAB).

The TI-RIPB approach to selecting LBEs is designed to ensure that an appropriate set of limiting events for each reactor technology are reflected in the selection of DBAs and that the full set of LBEs define the risk significant events for each design and technology. This is essential to ensure that risk insights are appropriately reflected in the design and licensing decisions.

The LBEs in each category are evaluated individually to support the tasks of assessing the performance of SSCs with respect to safety functions in response to initiating events and collectively to demonstrate that the integrated risk of a multi-module plant design meets the NRC Safety Goals.

There will be different LBEs for events affecting single and multiple reactor modules. An important outcome of the selection and evaluation of LBEs is to identify design features of the plant that are necessary and sufficient to ensure that risk goals are achieved and licensing requirements are met. The use of these insights in the derivation of performance requirements and principal design criteria for SSCs, including the radionuclide barriers is a topic of a future LMP deliverable on SSC safety classification.

- Implementation of the proposed TI-RIPB approach to selecting LBEs requires the development of deterministic and probabilistic inputs to the LBE selections that have sufficient technical adequacy to support such decisions.

  <u>LMP Approach</u>

  The approach to performing the required PRA inputs and for achieving the necessary technical adequacy of the PRA is the topic of a companion LMP deliverable to be provided for review. The PRA is introduced at an early stage of the design to support design decisions and the level of detail and scope of the PRA is consistent with the level of detail of the design and site characterization.

- The approach may be applied to advanced non-LWR plants with two or more reactor modules:

  <u>LMP Approach</u>

  In order to address the selection of LBEs for a plant with one or two or more reactor modules or radionuclide sources[16], the frequencies of LBEs are expressed in units of events per plant-year where a plant is defined as a collection of reactor modules within the scope of the license application[17]. Thus, each LBE may involve a plant response or release from one or multiple reactors or radionuclide sources The evaluation criteria on the frequency ranges for the LBE categories are as follows:

  - AOOs – event sequences with mean frequencies greater than 10^{-2} per plant-year

[16] Non-reactor sources include spent fuel storage and rad-waste process and storage systems.

[17] Each reactor module may be separately licensed, but when the second and subsequent modules are licensed the multi-module LBEs will be defined, and the plant capabilities to ensure that multi-module accident risks are not significant will be incorporated into the licensing basis.

- DBEs – event sequences with mean frequencies less than 10^{-2} per plant-year and greater than 10^{-4} per plant-year
- BDBEs – event sequences with mean frequencies less than 10^{-4} per plant-year and greater than 5×10^{-7} per plant-year.
- DBAs –are deterministically defined and are not selected on the basis of frequency. However, the plant response to each DBA corresponds to either a DBE, BDBE, or lower frequency sequence.

- Acceptable offsite dose evaluation criteria on the event sequence consequences for the LBE categories are defined by frequency-consequence evaluation criteria derived from Top Level Regulatory Criteria (TLRC). The TLRC frequency-consequence criteria are used to evaluate the risk significance of each LBE.

  LMP Approach

  Key elements of the TLRC used to develop the frequency-consequence criteria include:

  - AOOs – 10 CFR Part 20: 100 mrem total effective dose equivalent (TEDE) mechanistically modeled and realistically calculated at the exclusion area boundary (EAB). For the LMP facility, the EAB is expected to be the same area as the controlled area boundary.
  - DBEs – 10 CFR §50.34: 25 rem TEDE mechanistically modeled and realistically calculated at the EAB.
  - BDBEs – NRC Safety Goals for large release frequency and quantitative health objectives (QHOs) for the risk of individual fatality are mechanistically and realistically calculated out to 1 mile (1.6 km) from the site boundary for early health effects and 10 miles (16 km) from the site boundary for latent health effects.

- In addition to evaluating the risk significance of individual LBEs, the LMP approach to evaluating LBE includes several evaluation criteria to ensure that the integrated risk of the advanced non-LWR plant, which may be comprised of two or more reactor modules, is acceptably small and consistent with the NRC Advanced Reactor and Safety Goal policies.

  LMP Approach

  These criteria include:

  - The total frequency of exceeding of a site boundary dose of 100mrem shall not exceed 1/plant-year to ensure that the annual exposure limits in 10 CFR 20 are not exceeded.
  - The total frequency of a site boundary dose exceeding 750 rem shall not exceed 10^{-6}/plant-year. Meeting this criterion would conservatively satisfy the NRC Safety Goal Policy Statement [48] on limiting the frequency of a large release
  - The average individual risk of early fatality within 1 mile of the EAB shall not exceed $5x10^{-7}$/plant-year to ensure that the NRC Safety Goal QHO for early fatality risk is met
  - The average individual risk of latent cancer fatalities within 10 miles of the EAB shall not exceed $2x10^{-6}$/plant-year to ensure that the NRC safety goal QHO for latent cancer fatality risk is met.

- Coverage of the event and accident frequency spectrum.

  LMP Approach

  The frequency below which events are not selected as LBEs is 5×10^{-7} per plant-year. Satisfaction of the NRC safety goal QHOs is assured when this frequency is not exceeded. The PRA examines events to 10^{-8} per plant-year to assure that there are no "cliff edge effects" just below this *de minimis* frequency.

- Completeness in the types of events to consider:

  LMP Approach

  The kinds of events, failures, and natural phenomena that are evaluated include:

  - single, multiple, dependent, and common cause failures to the extent that these contribute to LBEs and their frequencies
  - events affecting one or more than one reactor module or radionuclide source within the scope of the license application
  - internal events (including transients and accidents) and internal and external plant hazards that occur in all operating and shutdown modes and potentially challenge the capability to satisfactorily retain any source of radioactive material.

- Treatment of Uncertainty

  LMP Approach

  Uncertainty distributions including upper and lower 95% confidence values are evaluated for the frequency and the consequence for each AOO, DBE, DBA and BDBE.

  - The mean frequency is used to determine whether the event sequence family is an AOO, DBE, or BDBE. If the upper or lower bound on the LBE frequency straddles two or more regions, the LBE is compared against the frequency and consequence criteria for each region.
  - Sources of uncertainty that are identified by the PRA and not fully resolved via quantification are addressed as part of a risk-informed evaluation of defense-in-depth as addressed in a companion LMP deliverable on defense-in-depth.
  - The mean consequences are explicitly compared to the consequence criteria in all applicable LBE regions.
  - The upper bound consequences for each DBA, defined as the 95%tile of the uncertainty distribution, shall meet the 10 CFR §50.34 dose limit at the EAB. Sources of uncertainty in both the frequencies and consequences of each LBE are identified and addressed in the LMP approach to defense-in-depth.

5. REFERENCES

[1] Idaho National Laboratory, “Next Generation Nuclear Plant Licensing Basis Event Selection White Paper”, INL/EXT-10-19521, September 2010, [ADAMS Accession No. ML102630246].

[2] 10 CFR Part 52.1, “Licenses, Certifications, and Approvals for Nuclear Power Plants – Definitions”, 2015

[3] NUREG-0800, Standard Review Plan, Chapter 15.0, “Introduction - Transient and Accident Analyses”, Revision 3 March 2007

[4] 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Code of Federal Regulations, Office of the Federal Register, 2007.

[5] 10 CFR 50.62, “Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants”, 2015

[6] 10 CFR 50.63, “Loss of all alternating current power”, 2015

[7] 10 CFR 50.44, “Combustible gas control for nuclear power reactors”, 2015

[8] 10 CFR 50.150, “Aircraft impact assessment”, 2015

[9] U.S. Nuclear Regulatory Commission, ‘Policy Statement on the Regulation of Advanced Reactors”; 73 FR 60612, October 14, 2008

[10] U.S. Nuclear Regulatory Commission, “Policy Statement on Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities” 60 FR 42622, August 16, 1995

[11] U.S. Nuclear Regulatory Commission, ‘Policy Statement on Safety Goals for Operation of Nuclear Power Plants”, 51 FR 28044 August 21, 1986

[12] U.S. Nuclear Regulatory Commission, “Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants”; 50 FR 32138, August 8, 1985

[13] SECY 2002-0076, “Semi-Annual Update of the Future Licensing and Inspection Readiness Assessment (FLIRA)”, May 8, 2002

[14] SECY 2003-0047, "Policy Issues Related to Licensing Non-Light Water Reactor Designs," U.S. Nuclear Regulatory Commission, March 28, 2003.SECY 2002-0139

[15] SRM 2003-0047, "Staff Requirements Memorandum for SECY 03-0047-Policy Issues Related to Licensing Non-Light Water Reactor Designs," U.S. Nuclear Regulatory Commission, June 26, 2003.

[16] SECY-2005-0006, “Second Status Paper on the Staff's Proposed Regulatory Structure for New Plant Licensing and Update on Policy Issues Related to New Plant Licensing”, January 7, 2005

[17] SECY-2010-0034, “Potential Policy, Licensing, and Key Technical Issues for Small Modular Nuclear Reactor Designs”, March 28, 2010

[18] SECY-2011-0079, “License Structure for Multi-Module Facilities Related to Small Modular Nuclear Power Reactors”, June 12, 2011

[19] NUREG-0800, Standard Review Plan, Chapter 19.0, “Probabilistic Risk Assessment and Severe Accident Evaluation For New Reactors”, Revision 3 December 2015

[20] SECY 2016-0012, “Accident Source Terms and Siting for Small Modular Reactors and Non-Light Water Reactors”, February 7, 2016

[21] NUREG-1860, “Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing”, U.S. Nuclear Regulatory Commission, December 2007.

[22] NUREG-2150, “A Proposed Risk Management Regulatory Framework”, U.S. Nuclear Regulatory Commission, April 2012

[23] U.S. Nuclear Regulatory Commission, “The Near-Term Task Force Review of Insights From the Fukushima Dai-Ichi Accident”, July 12, 2011

[24] NUREG/BR-0303, Guidance for Performance Based Regulation, U.S. Nuclear Regulatory Commission, December 2002

[25] ANS-53.1-2001, “Nuclear Safety Design Process for Modular Helium-Cooled Reactor Plants”, American Nuclear Society, December 2011

[26] Idaho National Laboratory, “Next Generation Nuclear Plant Defense-in-Depth Approach White Paper”, INL/EXT-09-17139, March 2009, [ADAMS Accession No. ML102630246].

[27] Department of Energy, “Yucca Mountain Repository Safety Analysis Report” Chapters 1.6 - 1.9 and 1.14, Docket No. 63–001, DOE/RW-0573, Rev. 1, February 2009

[28] NUREG-2108, “Technical Evaluation Report on the Content of the U.S. Department of Energy’s Yucca Mountain Repository License Application: Preclosure Volume: Repository Safety Before Permanent Closure”, Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, September 2011

[29] 10 CFR Part 63, “Disposal Of High-Level Radioactive Wastes In a Geologic Repository at Yucca Mountain, Nevada”, 2016

[30] NUREG-1804, “Yucca Mountain Review Plan”, Revision 2, July 2003

[31] ASME/ANS RA-Sb-2013, “Addenda to ASME/ANS RA-S–2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications”, American Society of Mechanical Engineers and American Nuclear Society, September 2013

[32] ASME/ANS RA S-1.4-2013, “Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants”, American Society of Mechanical Engineers and American Nuclear Society, December 2013

[33] 10 CFR Part 63.111, “Performance objectives for the geologic repository operations area through permanent closure”, 2015

[34] 10 CFR Part 63.112, “Requirements for preclosure safety analysis of the geologic repository operations area”, 2015

[35] SRM 90-16, “Staff Requirements Memorandum for SECY 90-16 Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements”, January 1990

[36] Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment i_n Risk-informed Decisions on Plant Specific Changes to the Current Licensing Basis," Federal Register, U.S. Nuclear Regulatory Commission, Revision 2, May 2011

[37] Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities," Federal Register, U.S. Nuclear Regulatory Commission , Revision 2, March 2009.

[38] International Atomic Energy Agency, “Safety Standards Series: Safety of Nuclear Power Plants Design Requirements”, NSR-1, September 2000

[39] United Kingdom Office of Nuclear Regulation, “Safety Assessment Principles for Nuclear Facilities”, 2014 Edition Revision 0

[40] U.S. Department of Energy, Preliminary Safety Information Document for the Standard MHTGR, DOE-HTGR-86-024, September 1988.

[41] U.S. Department of Energy, Probabilistic Risk Assessment for the Standard Modular High Temperature Gas-Cooled Reactor, DOE-HTGR-86-011, Revision 5, April 1988.

[42] NUREG-1338, "Draft Preapplication Safety Evaluation Report for the Modular High Temperature Gas-Cooled Reactor," U.S. Nuclear Regulatory Commission, March 1989.
[43] NUREG-1368, Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor, Final Report, U.S. Nuclear Regulatory Commission, February 1994
[44] Edward G. Wallace letter to NRC Document Control Desk, “PBMR White Paper: LBE Selection,” July 3, 2006.
[45] NRC letter to PBMR, “Requests For Additional Information Regarding Pebble Bed Modular Reactor (PBMR) Pre-Application White Papers,” September 24, 2007.
[46] PBMR Limited letter to NRC, “Response to Requests for Additional Information,” March 21, 2008.
[47] Exelon Generation Company Letter, Subject: Proposed Licensing Approach for the Pebble Bed Modular Reactor in the United States, January 31, 2002.
[48] U.S. Nuclear Regulatory Commission Letter, Subject: "NRC Staff's Preliminary Findings Regarding Exelon Generation's (Exelon's) Proposed Licensing Approach For The Pebble Bed Modular Reactor (PBMR)," March 26, 2002.See also SECY-95-0299, "Licensing Approach for Exelon's Pebble Bed Modular Reactor", December 19, 1995.
[49] INL/EXT-13-28205,” NRC Licensing Status Summary Report for NGNP”, Idaho National Laboratory, November 2014.
[50] Letter from J. Sam Armijo, Chairman U.S. Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards to Mr. R.W. Borchardt, Executive Director for Operations, U.S. Nuclear Regulatory Commission Letter, “Next Generation Nuclear Plant (NGNP) Key Licensing Issues”, May 15, 2013
[51] SECY 2011-0152, “Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors”, October 28, 2011
[52] 10 CFR Part 20, “Standards for Protection against Radiation Subpart C, Occupational Dose Limits”, 2016
[53] 10 CFR Part 20, “Standards for Protection against Radiation Subpart D, Radiation Dose Limits for Individual Members of the Public”, 2016
[54] 10 CFR Part 50, Appendix I, “Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion ‘As Low as is Reasonably Achievable’ for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents”, 2015
[55] 10 CFR Part 52, Licenses, Certifications, And Approvals For Nuclear Power Plants, Subpart C “Combined Licenses”, 2016
[56] 10 CFR Part 52.79, “Contents of applications; technical information in final safety analysis report”, 2015
[57] 40 CFR Part 190, “Environmental Radiation Protection Standards for Nuclear Power Operations”, 2016
[58] 10 CFR Part 100, “Reactor Site Criteria (Subpart B, Evaluation Factors for Stationary Power Reactor Site Applications on or After January 10, 1997)”:
[59] 10 CFR §50.34, “Contents of Applications: Technical Information - Radiological Dose Consequences”, 2015
[60] 10 CFR Part 50.34a, “Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors”, 2015
[61] 40 CFR Part 190.10, “Standards for normal operations”, 2016

[62] Idaho National Laboratory, "Next Generation Nuclear Plant Probabilistic Risk Assessment White Paper", INL/EXT-11-21270, Idaho National Laboratory, September 2011.
[63] NUREG-0800, Standard Review Plan, Chapter 15.0.3 "Design Basis Accident Radiological Consequence Analyses For Advanced Light Water Reactors", March 2007
[64] Farmer, F., "Reactor Safety and Siting: A Proposed Risk Criterion", *Nuclear Safety* 1967; pp. 539–48.
[65] NUREG-0800, Standard Review Plan, Chapter 15.0 "Introduction - Transient and Accident Analyses", Revision 3 March 2007
[66] American Nuclear Society, ANSI/ANS-53.1-2011, "Nuclear Safety Design Process for Modular Helium-Cooled Reactor Plants", December 21, 2011
[67] NEDO-33201, Revision 6, ESBWR Design Certification Probabilistic Risk Assessment, Chapter 10.4 COMPARISON OF RESULTS TO GOALS, (Adams Accession No. ML102880536)
[68] U.S. Department of Energy, Licensing Basis Events for the Standard MHTGR, DOE-HTGR-86-034, February 1987.
[69] GE Hitachi Nuclear Energy, "Final Scientific/Technical Report: Development/Modernization of an Advanced Non-LWR Probabilistic Risk Assessment," Federal Grant DE-NE0008325, 2017.
[70] Letter from Mario V. Bonaca, Chairman U.S. Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards to Honorable Nils Diaz, Chairman, U.S. Nuclear Regulatory Commission Letter, "Options and Recommendations for Policy Issues Related to Licensing Non-Light Water Reactor Designs", April 22, 2004
[71] SECY 2013-0029, "History of The Use and Consideration of the Large Release Frequency Metric By The U.S. Nuclear Regulatory Commission", March 22, 2013

APPENDIX A
REGULATORY FOUNDATION AND PRECEDENTS

Contents

Tables

Figures

APPENDIX A

A. REGULATORY FOUNDATION AND PRECEDENTS

There is a substantial set of prior activities, policies, practices and precedents stretching more than 30 years back in time that inform RIPB processes and uses. NRC and international regulations, policies, guidance, and other precedents that are relevant to the definition of LBEs and their treatment are discussed in this section. NRC and ACRS feedback on previous efforts to define LBEs for Advanced Non-LWRs are also reviewed for LBE definition guidance. This regulatory background is examined to investigate two aspects of the proposed TI-RIPB approach for the LMP project. The first is the process of defining and selecting the LBEs and the second is the development of the Top-Level Regulatory Criteria (TLRC) that are used to establish evaluation boundaries on the frequencies and radiological consequences for classifying and evaluating the LBEs. The scope of this review includes U.S. regulatory requirements as specified in the regulations, and supporting policies, Commission directives, regulatory guidance, and Standard Review Plan as well as international safety standards. Insights from NRC pre-licensing reviews of advanced non-LWRs are also included. This section of the white paper builds on the regulatory review in the NGNP White Paper on LBE selection [1] by incorporating more recent developments and precedents and by considering the need to have a reactor technology inclusive approach for selecting LBEs rather than one focused on HTGR-specific technology only. Observations and conclusions reached from this review that are used in the definition of the LBE approach are summarized at the end of this section.

A.1 U.S. Regulatory Foundation for the Selection of LBEs

A.1.1 NRC Requirements

This section reviews NRC requirements for insights on how to select LBEs for a new reactor design. This discussion reflects on the qualitative approach to risk used in the past, relying on judgment and prescription derived from years of LWR design, analysis and operations. The purpose is not to criticize, but rather to identify desirable attributes of a TI-RIPB approach to the selection of LBEs.

NRC regulatory requirements for the design of currently licensed and new reactors refer to several different kinds of events included within the licensing basis including anticipated operational occurrences (AOOs), design basis events (DBEs), postulated accidents, design basis accidents (DBA), and beyond design basis events (BDBE). The definitions of these events are similar to LBE types introduced in Section 1.4 however there are significant differences in licensing event terminology as shown in Table 1 .

For normal operations, including AOOs, the NRC regulations are, for the most part, generic and appear to generally apply to an advanced non-LWR plant. The applicant is required to classify the events considered within the design basis as either AOO or accident (DBA) based on a qualitative and presumably subjective assessment of the expected frequency of occurrence because there are no quantitative frequency criteria included. In many cases it is unclear whether the qualitative characterization of frequency refers to that for an initiating event or for an entire accident sequence. While the applicant's classification is subjected to NRC staff review there is no quantification of the event frequencies nor a prescribed method for ensuring that design specific events are adequately considered. A concern for advanced non-LWRs is that events that are uniquely appropriate for a given reactor technology are likely not represented on the supplied lists of generic LWR events, so it is necessary to have a method that is systematic and reproducible to derive the appropriate list of LBEs. For non-LWR plants whose designs depart in major ways from those of existing and even advanced non-LWRs, a more systematic and quantitative means of identifying the unique events and correctly classifying their frequencies would be necessary to ensure a safe design and contribute to a more predicable path to a license.

Table 1 Definitions of Licensing Basis Events

Event Type	NRC Definition	LMP Definition
Anticipated Operational Occurrences (AOOs)	*"Conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit*[1] *and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power."* [SRP 15.0 and 10 CFR 50 Appendix A]	Conditions of plant operation, events, and event sequences that are expected to occur one or more times during the life of the nuclear power plant which may include one or more reactor modules. Events and event sequences with frequencies of $1x10^{-2}$ per plant year and greater are classified as AOOs. AOOs take into account the expected response of all SSCs within the plant regardless of safety classification.
Design Basis Events (DBEs)	*"Conditions of normal operation, including AOOs, design-basis accidents, external events, and natural phenomena, for which the plant must be designed to ensure functions of safety-related electric equipment that ensures the integrity of the reactor coolant pressure boundary; the capability to shut down the reactor and maintain it in a safe shutdown condition; or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures."* [SRP 15.0]	Events and event sequences that are expected to occur one or more times in the life of an entire fleet of nuclear power plants, but less likely than an AOO. Events and event sequences with frequencies of $1x10^{-4}$ per plant year to $1x10^{-2}$ per plant year are classified as DBEs. DBEs take into account the expected response of all SSCs within the plant regardless of safety classification. The objective and scope of the DBEs to form the design basis of the plant is the same as in the NRC definition. However DBEs do not include normal operation and AOOs as defined in the NRC references.
Beyond Design Basis Event (BDBE)	"*This term is used as a technical way to discuss accident sequences that are possible but were not fully considered in the design process because they were judged to be too unlikely. (In that sense, they are considered beyond the scope of design-basis accidents that a nuclear facility must be designed and built to withstand.) As the regulatory process strives to be as thorough as possible, "beyond design-basis" accident sequences are analyzed to fully understand the capability of a design."* [NRC Glossary]	Events and event sequences that are not expected to occur in the life of an entire fleet of nuclear power plants. Events and event sequences with frequencies of $5x10^{-7}$ per plant year to $1x10^{-4}$ per plant year are classified as BDBEs. BDBEs take into account the expected response of all SSCs within the plant regardless of safety classification. The objective of BDBEs to assure the capability of the plant is the same as in the NRC definition.

[1] SRP 15.0 further breaks down AOOs into events with "moderate" frequency (events expected to occur several times during the plant life) and "infrequent" (events that may occur during the plant life)

Event Type	NRC Definition	LMP Definition
Design Basis Accidents (DBA)	*"Postulated accidents that are used to set design criteria and limits for the design and sizing of safety-related systems and components."* [SRP 15.0] "*A postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety*." [NRC Glossary and NUREG 2122]	Postulated accidents that are used to set design criteria and performance objectives for the design and sizing of SSCs that are classified as safety-related. DBAs are derived from DBEs and high consequence BDBEs based on the capabilities and reliabilities of safety related SSCs needed to mitigate and prevent accidents, respectively. DBAs are derived from the DBEs by prescriptively assuming that only SSCs classified as safety related are available to mitigate postulated accident consequences to within the 50.34 dose limits.
Licensing Basis Events (LBEs)	Term not used formally in NRC documents	The entire collection of events considered in the design and licensing basis of the plant, which may include one or more reactor modules. LBEs include normal operation, AOOs, DBEs, BDBEs, and DBAs

Moreover, establishing an appropriate set of reactor technology specific LBEs cannot wait until the submittal of a license application. This selection is essential to the development of any design and must be established very early in the design process.[2]

All the example events given in the definition of AOO in the regulations and in the supporting regulatory guides and Standard Review Plan [3] are applicable to LWRs. Many of these may not be applicable to a particular non-LWR design.

In the selection of LBEs it is expected that the selection will consider a comprehensive and exhaustive set of events from which to identify the "limiting" events. However specific criteria for how to determine which events are limiting are not provided in existing regulatory guidance. In addition, it is not clear from the regulatory guidance which events are considered to be limited by the selected events. This points to a need for a systematic and reproducible process to identify the DBAs for the deterministic safety analysis.

With few exceptions, such as provisions for protection against natural phenomena and inclusion of some generic events in the lists of example events such as loss of offsite power and station blackout, the regulations that have evolved for unplanned transients and accidents are light water reactor (LWR)-specific. The GDC define the types of design considerations that apply to the design of SSCs that prevent or mitigate a specified set of postulated accidents. For example, GDC typically indicate that safety systems must be able to perform their design basis functions given a single active failure and a concurrent loss of offsite power.

NRC's regulations do not have performance-based criteria to limit the consequences of BDBEs nor quantitative criteria for classifying events as BDBEs based on frequency other than noting they were considered too infrequent to be included in the design basis. In apparent response to events that have occurred but had not been anticipated in the original design and licensing bases, regulations have been added to provide protection against selected BDBEs. Examples of these include: anticipated transients without scram (ATWS) addressed in 10 CFR §50.62 [5], station blackout addressed in 10 CFR §50.63.

The regulations associated with licensing events and their supporting regulatory documents do not distinguish well between events and event sequences for the purpose of characterizing the frequency of occurrence and classifying as either an AOO, DBA, or BDBE. The term "sequence of events" is referred to here in the context of analyzing how the plant responds to initiating events. The point here is a given event may be characterized at a certain frequency level and severity of plant impact, but when compounded by additional failures both the frequency and the level of impact are different. Hence, there may be different LBEs having different levels of frequency and severity stemming from the same initiating event. In reviewing the regulatory documents, it is extremely difficult to sort out in most cases whether the term "events" refer to initiating events only or to some sequence of events. A goal of the LMP is to consider initiating events and the associated event sequences as distinct challenges to the safety functions in order to provide sufficient completeness in the identification of LBEs.

In many cases the events classified as AOOs or DBAs as discussed in the regulations and supporting SRP are referred to as "initiating events". By applying the single failure criterion, the safety analysis for the DBAs includes the requirement that the "worst" active single failure be assumed in demonstrating that

2 One additional definition is required to understand the importance of the terms. The *Design bases* means that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state of the art" practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals. [10CFR50.2 Definitions]

safety criteria are met, however the probability of the single failure does not have any bearing on the classification of the event. In addition, non-safety related SSCs including offsite power supplies are assumed not to be available in the deterministic safety analysis of DBAs which also would be considered if the frequency of the DBA were to be assessed.

With few exceptions, there does not appear to be a consideration of the probability of a common cause failure that could occur in combination with an initiating event to produce a DBA even though the service experience indicates that there have been many occurrences of such events. The application of the single failure criterion for DBAs seems to assume that common cause failures will be prevented by meeting the design requirements. In the limited cases of selected BDBE requirements such as those for anticipated transients without scram ATWS and station blackout (SBO), event sequences that could be caused in part by a common cause failure and involve multiple failures of redundant components are identified as being comprised of a sequence initiated by an AOO (transient for ATWS and loss of offsite power for SBO). However, a systematic way to consider both events and event sequences that could be comprised of combinations of single failures and common cause failures is not included in the enumeration of prescriptive events nor in the characterization of their frequencies or level of severity of the challenge.

An important insight from this review is that, based on what can be gleaned from the regulations and supporting documents, the historical approach to selecting an appropriate set of LBEs for a given design is ad hoc. The challenge facing designers and licensees for advanced non-LWRs is to find a process for selecting LBEs that is systematic, reproducible, and capable of identifying the appropriate limiting events for a given design.

A.1.2 NRC Policy Statements

Advanced non-LWR designs need to adhere to relevant NRC policies. Each of the reviewed policies was examined for insights on the desirable attributes for an effective approach to select an appropriate set of LBEs for an advanced non-LWR.

A.1.2.1 Advanced Reactor Policies

Advanced non-LWR designs that may benefit from the LMP licensing strategy as expressed in this paper and associated documents are implementing design features identified in the NRC’s Advanced Reactor Policy which was revised in 2008 [9]. For advanced non-LWR reactor designs, the NRC expects at least the same degree of protection of the environment, public health and safety, and common defense and security that is required for current generation LWRs. The NRC also expects that advanced reactors will provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions.

This Advanced Reactor Policy does not offer or refer to any guidance as to how LBEs will be defined for such reactors. It is reasonable to assume that reactors that rely more on passive and inherent safety features and less on active systems, use different materials for reactor fuel, coolant, and moderator, and have different design and configuration for radionuclide barriers may have limiting LBEs that are unique and specific relative to those previously defined for LWRs.

In 1995 the NRC issued its policy on the use of PRA methods in the regulatory process [10]. The essence of this policy is reflected in the following statement:

> *This statement reflects the policy that the Nuclear Regulatory Commission (NRC) will follow in the use of probabilistic risk assessment (PRA) methods in nuclear regulatory matters. The Commission believes that an overall policy on the use of PRA methods in nuclear regulatory activities should be established to that the many potential applications of PRA can be implemented in a consistent and predictable manner that would promote regulatory stability and efficiency. In addition, the Commission believes that the use of PRA technology in NRC regulatory activities*

should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach…."

A key objective of the LBE selection approach described in this paper is to have an approach that is consistent with this policy statement which clearly articulates the need for a risk-informed approach.

Another key policy that is essential to help implement risk-informed decision making is the NRC Safety Goal Policy [12], whose purpose is summarized in the following statement.:

> *This policy statement focuses on the risks to the public from nuclear power plant operation. Its objective is to establish goals that broadly define an acceptable level of risk. In developing the policy statement, the NRC sponsored two public workshops during 1981, conducted a 2 year evaluation during 1983 to 1985, and received the views of its Advisory Committee on Reactor Safeguards.*

This safety goal policy established two qualitative safety goals, supported by two quantitative health objectives bases on the principle that nuclear risks should be a small fraction of other societal and individual risks. This policy reflects the NRC's judgment on the question of "how safe is safe enough?" and provides goals and criteria which are used in Section 3 of this report to evaluate the risk significance of LBEs and the overall plant risks resulting from an entire collection of LBEs.

SECY 2002-0076, "Semi-Annual Update of the Future Licensing and Inspection Readiness Assessment," [13] described Exelon's proposed licensing approach for the pebble bed HTGR design. Exelon proposed conformance with current regulations but recognized that many of the regulatory requirements were based on LWR technology. A risk-informed process would be employed to define plant design events, acceptance criteria, and SSCs. In its preliminary evaluation, NRC staff concluded that the proposed licensing approach, if adequately implemented, was a reasonable process for ensuring that the Commission's regulations would be met and for identifying pebble bed modular reactor (PBMR)-specific regulatory requirements.

SECY 2003-0047, "Policy Issues Related to Licensing Non-Light Water Reactor Designs" [14] offers staff recommendations on several relevant policy issues that had been originally defined in an earlier policy statement (SECY 2002-0139). Of these issues, Issue 4, "Use of PRA to Support Licensing Basis," specifically relates to the treatment of LBEs and is discussed herein. The Staff Requirements Memorandum (SRM) for SECY 2003-0047 [15] stated the Commissioners approval of the staff recommendations on this issue.

With respect to Issue 4, the staff recommended that the Commission take the following actions:

> Modify the Commission's guidance, as described in the SRM of July 30, 1993, to put greater emphasis on the use of risk information by allowing the use of a probabilistic approach in the identification of events to be considered in the design, provided there is sufficient understanding of plant and fuel performance and deterministic engineering judgment is used to bound uncertainties.
>
> Allow a probabilistic approach for the safety classification of structures, systems, and components.
>
> Replace the single failure criterion with a probabilistic (reliability) criterion.

This recommendation is consistent with a risk-informed approach. It should be noted that this recommendation expands the use of probabilistic risk assessment (PRA) into forming part of the basis for licensing and thus puts greater emphasis on PRA quality, completeness, and documentation.

Also included, but left unresolved from the issues of SECY 2003-0047, were policy issues associated with the treatment of integrated risk on multi-module plants and for modular reactor designs, which are part of Issue 1 as stated in the SECY.

In its SRM to SECY-2003-047 the Commission asked the staff to identify options and provide more specifics on the treatment of integrated risks for multi-module plants. In response the staff issued SECY 2005-0006 [16], in which the staff recommended that the integrated risks of multi-module plants be addressed. In characterizing the risks from a multi-module plant, the staff noted that two different types of event sequences need to be considered:

> *"It should also be noted that in assessing the risk from plants consisting of multiple reactor modules, the event sequences that contribute to risk will generally fall into two basic categories (1) those that affect each reactor module individually and (2) those that can affect two or more modules simultaneously (e.g., seismic events). Accordingly, the overall risk from a plant comprised of multiple reactor modules consists of the sum of the risk from both categories, and may be lower then the sum of the risk from all modules if they were treated separately, particularly if some systems are shared among reactor modules. This would be due to the fact that the risk from event sequences that affect all reactor modules simultaneously may not be equal among the reactor modules."*

In this SECY the staff recommended that advanced non-LWR plants with multi-module designs assess the integrated risk of the facility according to an Option 3 with consideration of both frequency and power level of the reactor modules. The other two options considered were an Option 1 in which integrated risk is not considered, and an Option 2 in which the integrated risk would be included by considering the frequency of both single and multiple reactor accidents but not addressing the power level in the assessment of consequences:

> *"On this basis, the staff has developed a proposed position endorsing Option 3. Option 3 realistically accounts for modular reactor characteristics by treating accident prevention independent of reactor power, while allowing the assessment of accident mitigation risk measures to consider reactor power, thus not imposing a de facto more stringent goal than implied by the Safety Goal Policy. In addition, Option 3 would be most consistent with the proposed Energy Bill language that would allow a set of reactor modules to be treated as a single unit for the purposes of financial protection (i.e., the risk from the set of reactor modules should not exceed that from a single large reactor).Option 3 would result in staff treatment of the risk associated with modular reactors as follows:*
>
> - *taking into consideration the integrated effect of risk when assessing accident prevention for modular reactor designs, independent of reactor power level, and*
> - *taking into consideration the integrated effect of risk when assessing accident mitigation for modular reactor designs in a fashion that allows for consideration of the effect of reactor power level."*

In 2006 the Commission in its SRM to SECY 06-007 approved the NRC staff's recommendation to issue an Advanced Notice of Proposed Rulemaking (ANPR) on approaches for making technical requirements for power reactors risk-informed, performance-based, and technology neutral, subject to the comments and edits provided in the SRM. The staff recommendation identified a number of policy issues including the question of the level of safety to be required for advanced reactors and the question of the integrated risk of multiple module plants. The Commission approved the staff's recommendation to supplement the

ANPR with new information, as needed. Subsequently, based on industry feedback that it was premature to propose rulemaking for advanced reactors, a revised recommendation was made in SECY-07-0101 to defer rulemaking until there was a license application for a PBMR or NGNP reactor. This recommendation was approved by the Commission in the SRM to SECY 07-0101, and hence policy issues for licensing advanced non-LWRs dating back to SECY 2003-0047 were never resolved.

The appreciation that accident sequences may involve two or more reactors has continued to evolve and was certainly manifested during the Fukushima Daiichi accident that produced core damage and containment breach on three reactor units.

A.1.2.2 NRC Small Modular Reactor Precedents

Because most advanced non-LWRs will employ a modular reactor design approach, a review of NRC policies for licensing SMRs will be beneficial in developing a suitable LBE approach.

SECY 2010-0034, "Potential Policy, Licensing, and Key Technical Issues for Small Modular Nuclear Reactor Designs," [17] identifies a number of potential policy and licensing issues based on the preliminary design information provided by pre-applicants and discussions with the designers and DOE regarding their proposed approaches to addressing key issues. Issues included accident selection for small modular reactors. With respect to this issue, the staff noted:

> *"In the August 2008 NGNP Licensing Strategy, the Commission stated that licensing-basis event categories (i.e., abnormal occurrences, design-basis accidents, and beyond-design basis accidents) would be established based on the expected probability of event occurrence. However, selection of licensing basis events within each category would be performed using deterministic engineering judgment complemented by insights from the NGNP PRA. In general, the NRC staff expects to apply this approach to all SMRs.*
>
> *Although identification of many accident scenarios will likely be straightforward, the application of certain scenarios may require Commission consideration. For example, designers of HTGRs have previously proposed that the failure of the vessel or piping connecting the reactor vessel and steam generator vessel need not be considered as a design basis event. In addition, although the Commission has previously stated that certain events should be addressed for non-LWR designs, subsequent research and evaluations may challenge the need to analyze these low probability events."*

The announced NRC staff plans to develop proposed resolutions to the issues in SECY 2010-0034 by continuing to obtain information from DOE, potential design and license applicants, and other sources; identifying and developing proposals for the resolution of policy issues; and where appropriate, preparing papers proposing resolutions of these issues with recommendations for consideration and approval by the Commission. This approach is a reasonable model for non-LWRs generally, whether large or small.

In SECY-16-0012 [20] the NRC staff recommended that design specific DBA mechanistic source terms can be used to address siting and emergency planning requirements for SMRs including those based on non-LWR technology. Importantly this SECY recommends that design specific DBA source terms used be based on accidents involving a single module:

> *The siting dose criteria are expected to be evaluated through DBA dose analyses on a per-reactor basis, even for multi-module plants. This is because of the design protection against external events that may affect more than one module concurrently, separation and independence of the modules' systems, structures and components and safety functions, and design against common cause failures among modules, in accordance with GDCs 2, 4, and 5.*

This means that the siting of a multi-module plant, including the determination of the EAB, LPZ and population center distances, is currently expected to be based upon the evaluation of a single reactor.

This part of the precedent review is helpful in identifying the need for and potential benefits of a risk-informed approach that is capable of identifying an appropriate set of LBEs for advanced non-LWRs employing a modular design approach. Developing the means of evaluating multi-module configurations and other onsite non-reactor risk also provides flexibility to designers regarding how to best configure the plant for economic as well as safety considerations. In addition it is necessary to incorporate appropriate ground rules for development and review of reactor-specific and scenario specific source terms.

A.1.3 NRC Guidance

A.1.3.1 Standard Review Plan (NUREG-0800)

NUREG-0800, "Standard Review Plan (SRP) Chapter 15.0," [3] identifies the types of AOOs and DBAs that must be postulated for LWRs. Hence, its review is relevant to developing a process for selecting LBEs.

SRP Chapter 15.0 includes a listing of generic LWR events given as example events and the applicant is directed to propose the design specific AOOs and DBAs along with the design description for the NRC staff to review. There was no guidance or acceptance criteria identified that would ensure that the "limiting" AOOs and DBAs appropriate and applicable to the design are selected for the deterministic safety analysis. A design specific version of the SRP has been developed for the NuScale SMR but the corresponding Section 15.0 does not indicate or refer to a method to identify DBAs or AOOs that are specific for that design nor acceptance criteria to ensure that the "limiting" events have been selected. There is no comparable guidance included for non-LWRs.

In Section 19.0 of the SRP [19] includes several sections of regulatory guidance for the technical adequacy of the PRA required to be part of the license application that appear to be relevant to advanced non-LWR designs that employ passive safety features and modular reactor designs. These sections offer limited guidance for the probabilistic inputs to selecting LBEs.

Regarding PRAs performed for passive designs the following review guidance is provided.

"Design-Specific PRA (Procedures Specific to Passive Designs)

1. *The issue of T-H uncertainties in passive plant designs arises from the passive nature of the safety-related systems used for accident mitigation. Passive safety systems rely on natural forces, such as gravity, to perform their safety functions. Such driving forces are small compared to those of pumped systems, and the uncertainty in their values, as predicted by a best-estimate T-H analysis, can be of comparable magnitude to the predicted values themselves. Therefore, some accident sequences with a frequency high enough to impact results, but not predicted to lead to core damage by a best- estimate T-H analysis, may actually lead to core damage when PRA models consider T- H uncertainties. One approach to addressing this issue is to perform sensitivity studies to see the effect of assuming bounding values for T-H parameters on success criteria and performing studies of the sensitivity of changes in success criteria on CDF.*
   A. *The reviewer assures that the applicant has (1) identified all key T-H parameters that could affect the reliability of a passive system and introduce uncertainty into the determination of success criteria, and (2) accounted for the uncertainty in the analyses that establish the success criteria.*

B. *The reviewer examines the results of any sensitivity studies performed by the applicant and the choice of T-H accident analysis codes used to perform such studies. Applicants frequently use the Modular Accident Analysis Program (MAAP) code for such studies. The staff is aware of T-H modeling issues with the code that could compromise its ability to confirm the validity of the PRA success criteria involving minimal sets of mitigating equipment. Use of this code is acceptable only if sufficient benchmarking studies have been done which compare MAAP results with those of a T-H code the staff has reviewed and approved and show that MAAP is able to capture the important T-H phenomena and the timing of such phenomena in simulations of accident sequences included in the PRA. If a small set of accident scenarios is used in the studies, the reviewer confirms that the applicant has provided an adequate rationale for its selection of scenarios, including a discussion of the criteria used for selection.*

2. *For passive plant designs, the staff reviews the applicant's use of the PRA to identify "nonsafety-related," SSCs that require regulatory treatment (i.e., to support the RTNSS program). Specifically this includes the following evaluations performed by the applicant as described in SRP 19.3:*
   A. *Evaluation of the risk significance of nonsafety systems using the Focused PRA*
   B. *Evaluation of uncertainties associated with assumptions made in the PRA models of passive systems*
   C. *PRA initiating event frequency evaluation*

Although this guidance was developed for LWR designs, it provides useful guidance for PRA reviews for advanced non-LWRs that employ passive safety features. Because the LMP approach to selecting LBEs is based in part on information derived from the PRA, consideration of the response of passive safety features to LBEs will be an important element of the LBE selection process for non-LWRs that employ passive safety features.

The SRP for Chapter 19 also includes the following guidance for PRA reviews for modular integral pressurized water reactors that employ modular reactor designs.

Design-Specific PRA (Procedures Specific to Integral Pressurized Water Reactors)

1. *For small, modular integral pressurized water reactor designs, the staff reviews the results and description of the applicant's risk assessment for a single reactor module; and, if the applicant is seeking approval of an application for a plant containing multiple modules, the staff reviews the applicant's assessment of risk from accidents that could affect multiple modules to ensure appropriate treatment of important insights related to multi-module design and operation.*

   The staff will verify that the applicant has:

   - *Used a systematic process to identify accident sequences, including significant human errors, that lead to multiple module core damages or large releases and described them in the application*
   - *Selected alternative features, operational strategies, and design options to prevent these sequences from occurring and demonstrated that these accident sequences are not significant contributors to risk. These operational strategies should also provide reasonable assurance that there is sufficient ability to mitigate multiple core damages accidents.*

The above guidance is relevant to advanced non-LWR designs that include modular reactors for the PRA inputs to selecting LBEs.

A.1.3.2 NUREG-1860 Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing [21]

NUREG-1860 is clearly the most relevant document that has been identified in providing a possible methodology for selection of LBEs as well as for formulation of reactor specific regulatory requirements for advanced non-LWRs. The citations in this paper are focused on event sequence identification and derivation of design bases events for specialized evaluation. Other topics relevant to different topics will be included in subsequent LMP papers.

The objectives of the framework include:

> ***Risk-informed*** *- Ensure that risk information and risk insights are integrated into the decision making process such that there is a blended approach using both probabilistic and deterministic information.*
>
> ***Performance-based*** *- When implemented, the guidance and criteria produce a set of safety requirements that are based on plant performance, and do not use prescriptive means for achieving its goals.*
>
> ***Defense-in-depth*** *- Defense-in-depth is an integral part of the framework such that uncertainties are accounted for in the requirements for design, construction, and operation.*
>
> ***Flexible*** *- The framework should, allow the licensing process to support reactors of diverse designs and be developed in such a manner that, as new information and knowledge are gained, changes to the regulatory structure can be implemented effectively and efficiently.*

The framework was developed from the top-down starting with the Atomic Energy Act and includes the following elements:

> ***Element 1: Goals and Expectations****: These start with the Atomic Energy Act principle of providing adequate protection of public health and safety and NRC's expectations for safety security and preparedness. Safety expectations are anchored to the Safety Goal policy and NRC's expectations that advanced reactors will provide enhanced margins of safety and comply with the Safety Goal policy with the QHOs representing the level of safety intended to achieve. NRC's security expectations are that advanced reactors will provide enhanced margins of safety and utilize simplified, inherent, passive or other innovative means to accomplish their safety and security functions.*
>
> ***Element 2: Defense-In-Depth:*** *A core principle of the NRC's safety philosophy has always been the principle of defense-in-depth. This principle remains basic to the safety, security, and preparedness expectations in the framework. The ultimate purpose of defense-in-depth is to compensate for uncertainty (e.g., uncertainty due to lack of operational experience with new technologies and new design features, uncertainty in the type and magnitude of challenges to safety). In licensing future reactors, the treatment of uncertainties will play a key role in ensuring that safety limits are met and that the design is robust for unanticipated factors.*
>
> ***Element 3: Safety Fundamentals:*** *This element provides the path, or process, from the high level goals and expectations to actually establishing specific requirements…. The process chosen to initially identify and define the requirements and regulations needs to implement the safety, security, and preparedness expectations and ensure protection of the public health and safety.*

Safety fundamentals have been defined, using a defense-in-depth approach, in the form of protective strategies that, if met, will ensure the protection of the public health and safety with a high degree of confidence…. A top-down analysis of each protective strategy leads directly to a categorization of the kinds of requirements that can ensure that the protective strategies are met.

Element 4: Licensing Basis: *A major goal is that the regulatory licensing basis be risk-informed…. The current regulatory structure is deterministic and is being modified in places to incorporate risk insights. A risk-informed regulatory structure should integrate risk from conception…In the framework, probabilistic criteria integrated with deterministic criteria based on plant specific considerations are used to establish potential new requirements….In using a probabilistic process, confidence in the technical acceptability becomes a key factor. Therefore, the technical acceptability of the PRA is part of this element.*

The licensing basis criteria are parallel and complementary with the Protective Strategies, in support of the NRC's defense-in-depth expectations, as shown in Figure 2-5. The probabilistic criteria include compliance with the quantitative health objectives (QHOs) of the NRC's safety goals.

The framework establishes probabilistic criteria to ensure that:

- *The integrated plant risk is acceptable in terms of the QHOs of the NRC's safety goal policy statement,*
- *A frequency consequence (F-C) curve is developed …that together with the plant PRA is used to select licensing basis events.*
- *The selection of those events that are used to establish the licensing basis of the design (licensing basis events or LBEs) is carried out in a risk-informed manner,*
- *The LBEs meet the F-C curve with margin, and*
- *The safety classification of systems, structures, and components (SSCs) reflects their importance in reducing plant risk.*

In selecting both the LBEs and the safety significant SSCs, defense-in-depth measures are incorporated, but, in addition, the risk information from the PRA is used to focus attention on the risk-significant aspects of the design.

LBEs derived from the PRA need to meet stringent probabilistic acceptance criteria and, depending on their frequency, need to meet additional deterministic (defense-in-depth) criteria. In this manner, the LBEs provide additional assurance that the design has adequate defense-in-depth in the form of sufficient margin to account for uncertainties. The LBEs also include a deterministically selected event, used in assessing site suitability.

Element 5: Integrated Process: *The process for identifying the requirements begins with the protective strategies. Each one is examined with respect to what are the various threats or challenges that could cause the strategy to fail. These challenges and threats are identified using a logic tree to perform a "systems analysis" of the strategy to identify potential failures. The defense-in-depth principles are then applied to each protective strategy. Defense-in-depth measures are identified which should be incorporated into the requirements to help prevent protective strategy failure. This approach forms the process for the selection of "topics." Requirements are then identified for each topic.*

Part of the process involves development of guidance to be used for actually writing the requirements. This guidance addresses writing the requirements in a performance-based fashion, *incorporating lessons learned from past experience, and utilizing existing requirements and guidance, where practical. The guidance also ensures that the probabilistic process for establishing the licensing basis are incorporated. All of the above are integrated and results in a set of potential requirements which serve to illustrate and establish the feasibility of developing a risk-informed and performance-based licensing approach.*

The framework set forth in NUREG-1860 has influenced the LMP selection approach for selecting LBEs for advanced non-LWRs. Our approach differs from that of NUREG-1860 in some important instances as well. This is discussed more fully in Section 3 of this paper.

advanced non-LWRs and provides recommendations for PRA technical adequacy requirements for PRAs used to support the licensing bases. This aspect of the document is reviewed in the companion LMP paper on PRA.

A.1.3.3 NUREG-2150 A Proposed Risk Management Regulatory Framework [22]

In early 2011, an NRC Commissioner led a Risk Management Task Force (RMTF) to evaluate how the agency should be regulating 10 to 15 years in the future. The RMTF was chartered:

"to develop a strategic vision and options for adopting a more comprehensive and holistic risk-informed, performance-based regulatory approach for reactors, materials, waste, fuel cycle, and transportation that would continue to ensure the safe and secure use of nuclear material."

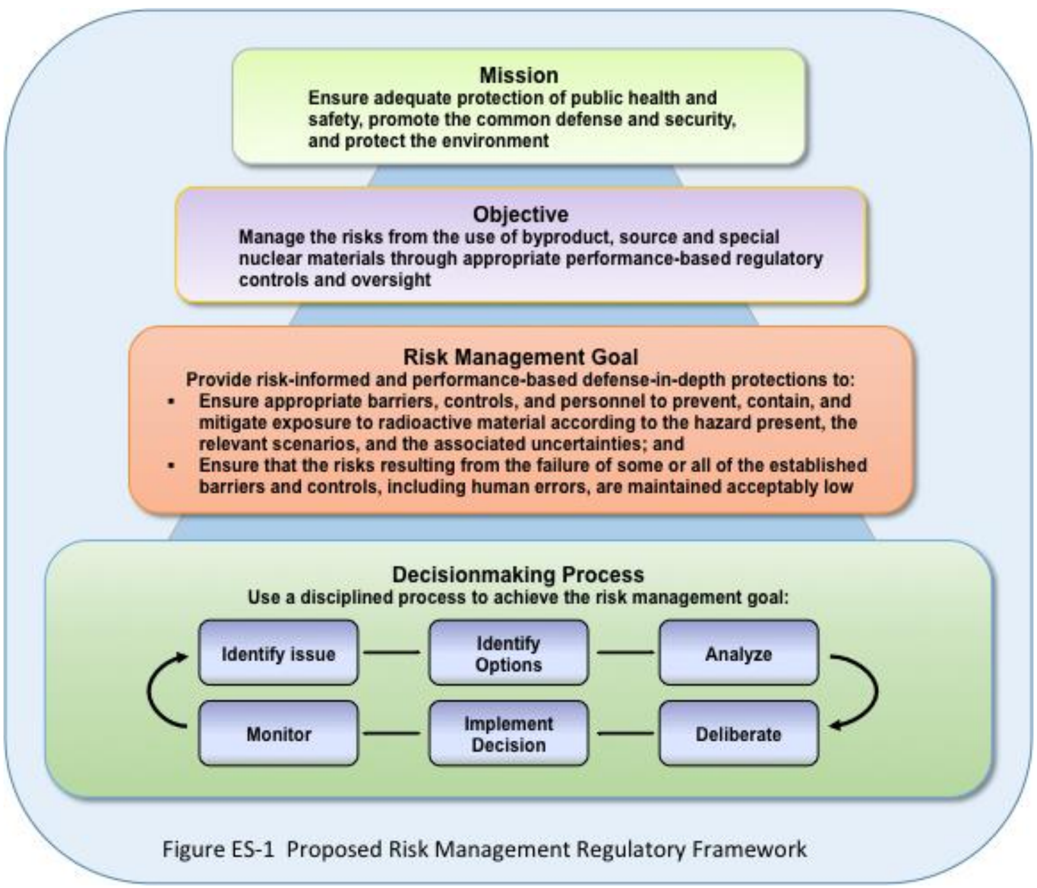

Figure ES-1 Proposed Risk Management Regulatory Framework

Figure 1 Regulatory Framework Proposed in NUREG-2150

The RMTF report developed a series of findings and recommendations covering the full scope of nuclear facilities including existing reactors and fuel cycle facilities and advanced non-LWRs to be licensed in the future. One of the key findings was that risk management should be stated as the NRC's objective according to a proposed framework shown in **Figure** 1 (reproduced from Figure ES-1 in NUREG-2150). The first recommendation was that:

> *The NRC should formally adopt the proposed Risk Management Regulatory Framework through a Commission Policy Statement.*

Specific recommendations were made for implementing the Framework in Figure 2-1 for Generation IV reactors which encompass the advanced non-LWR designs intended for the LBE selection process outlined in this white paper. Those recommendations are listed in **Table** 2. These recommendations were considered in the development of the LBE selection process proposed in Section 3 of this paper.

Table 2 NRC Risk Management Task Force Recommendations for Generation IV Reactors

RMTF Recommendation for Generation IV Reactors	Referenced RMTF Recommendation for Power Reactors
GIV-R-1: For Generation IV reactors, the RMTF recommends that the concept of design-basis accidents be maintained, but the NRC should be amenable to and promote, where practical, the adoption of more risk-informed approaches for the selection of relevant scenarios (e.g., alternatives to the single failure criterion) for design-basis accidents.	N/A
GIV-R-2: Apply Recommendation PR-R-2 (design-enhancement category) to Generation IV reactors.	PR-R-2: The NRC should establish through rulemaking a *design-enhancement category* of regulatory treatment for beyond-design-basis accidents. This category should use risk as a safety measure, be performance-based (including the provision for periodic updates), include consideration of costs, and be implemented on a site-specific basis.
GIV-R-3: Apply Recommendation PR-R-3 (include external events in design-enhancement category) to Generation IV reactors.	PR-R-3: The NRC should reassess methods used to estimate the frequency and magnitude of external hazards and implement a consistent process that includes both deterministic and PRA methods. Consideration of the risks from beyond-design-basis external hazards should be included in the design-enhancement category described in Recommendation PR-R-2.
GIV-R-4: Apply Recommendation PR-R-4 (periodically evaluate new information regarding external hazards) to Generation IV reactors.	PR-R-4: The NRC should establish a program to systematically collect, evaluate, and communicate external hazard information.
GIV-R-5: Apply Recommendation PR-R-5 (issue guidance to adopt risk-informed and performance-based defense-in-depth) to Generation IV reactors.	PR-R-5: The NRC should apply the risk-informed and performance-based defense-in-depth concept to power reactors in a more quantitative manner.
GIV-R-6: Apply Recommendation PR-R-6 (develop guidance and consistent approach between safety and security) to Generation IV reactors.	PR-R-6: The NRC should develop and implement guidance for use in its security regulatory activities that uses a common language with safety activities and harmonizes methods with risk assessment and the proposed risk-informed and performance-based defense-in-depth framework.

[what does the LMP intend to do with these recommendations These recommendations are consistent with the LMP goal of using a TI-RIPB approach to making licensing decisions for advanced non-LWRs, including the selection of LBEs and application of defense-in-depth principles that span the full spectrum hazards, frequencies, and consequences.

A.1.3.4 Fukushima Accident and NRC Near-Term Task Force (NTTF) Report [23]

On March 11, 2011, the Great East Japan Earthquake occurred 231 miles northeast of Tokyo off the coast of Honshu Island. This earthquake resulted in the automatic shutdown of 11 nuclear power plants at four sites along the northeast coast of Japan (Onagawa 1, 2, and 3; Fukushima Dai-ichi 1, 2, and 3; Fukushima Dai-ni 1, 2, 3, and 4; and Tokai 2). The earthquake precipitated a large tsunami that is estimated to have exceeded 14 meters (45 feet) in height at the Fukushima Dai-ichi Nuclear Power Plant site. The earthquake and tsunami produced widespread devastation across northeastern Japan, resulting in approximately 25,000 people dead or missing, displacing many tens of thousands of people, and significantly impacting the infrastructure and industry in the northeastern coastal areas of Japan. The site inundation at Fukushima Daiichi led to a severe core damage accident on three of the six reactor units, and containment breach on at least one unit, due to a prolonged loss of AC and DC power due to flood damage to onsite emergency diesel-generators and electrical switchgear. The management of the accident, which was successful in preventing significant off-site radiological exposures to the public, was complicated by the concurrent needs to manage the accident and to protect the remaining units and associated spent fuel storage facilities.

In July 2011, the NRC issued its Near Term Task Force (NTTF) Report with recommendations for enhancing reactor safety. It included the greater use of risk considerations and defense-in-depth in its final report. There are 12 specific recommendations are organized into the following topics.

Clarifying the Regulatory Framework

1. *The Task Force recommends establishing a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations. (Section 3)*

Ensuring Protection

2. *The Task Force recommends that the NRC require licensees to reevaluate and upgrade as necessary the design-basis seismic and flooding protection of structures, systems, and components for each operating reactor. (Section 4.1.1)*
3. *The Task Force recommends, as part of the longer term review, that the NRC evaluate potential enhancements to the capability to prevent or mitigate seismically induced fires and floods. (Section 4.1.2)*

Enhancing Mitigation

4. *The Task Force recommends that the NRC strengthen station blackout mitigation capability at all operating and new reactors for design-basis and beyond-design-basis external events. (Section 4.2.1)*
5. *The Task Force recommends requiring reliable hardened vent designs in boiling water reactor facilities with Mark I and Mark II containments. (Section 4.2.2)*
6. *The Task Force recommends, as part of the longer term review, that the NRC identify insights about hydrogen control and mitigation inside containment or in other buildings as additional information is revealed through further study of the Fukushima Dai-ichi accident. (Section 4.2.3)*
7. *The Task Force recommends enhancing spent fuel pool makeup capability and instrumentation for the spent fuel pool. (Section 4.2.4)*

8. *The Task Force recommends strengthening and integrating onsite emergency response capabilities such as emergency operating procedures, severe accident management guidelines, and extensive damage mitigation guidelines . (Section 4.2.5)*

Strengthening Emergency Preparedness

9. *The Task Force recommends that the NRC require that facility emergency plans address prolonged station blackout and multiunit events. (Section 4.3.1)*
10. *The Task Force recommends, as part of the longer term review, that the NRC pursue additional emergency preparedness topics related to multiunit events and prolonged station blackout. (Section 4.3.1)*
11. *The Task Force recommends, as part of the longer term review, that the NRC should pursue emergency preparedness topics related to decisionmaking, radiation monitoring, and public education. (Section 4.3.2)*

Improving the Efficiency of NRC Programs

12. *The Task Force recommends that the NRC strengthen regulatory oversight of licensee safety performance (i.e., the Reactor Oversight Process) by focusing more attention on defense-in-depth requirements consistent with the recommended defense-in-depth framework. (Section 5.1)*

These recommendations identify the need to address the risk of multiunit accidents via enhancements to emergency planning, ensure adequate protection for seismic events and external flooding, and to provide the capabilities for coping with an extended loss of AC power. These recommendations are supported by the NTTF finding recognizing the complementary roles of defense-in-depth and PRA.

> *"The Task Force finds that the Commission's longstanding defense-in-depth philosophy, supported and modified as necessary by state-of-the-art probabilistic risk assessment techniques, should continue to serve as the primary organizing principle of its regulatory framework. The Task Force concludes that the application of the defense-in-depth philosophy can be strengthened by including explicit requirements for beyond-design-basis events."*

general guidance on the importance of a modern RIPB framework and the importance of determining a well-structured and set of LBEs to base design conditions for SSCs.

A.1.3.5 NRC Guidance for Performance Based Regulation [24]

An objective for the LMP is to define an approach to selecting LBEs that is both risk-informed and performance-based (RIPB). Regulatory documents reviewed in previous sections contain useful guidance for how LBEs may be risk-informed but do not explicitly define what additional characteristics may be necessary to classify the approach as performance-based. Guidance on criteria for defining performance based alternatives in regulatory decision making is found in NUREG/BR-0303 [24].

This That document provides guidance on a process for developing a performance-based alternatives to more prescriptive approaches for regulatory decision making. The U.S. Nuclear Regulatory Commission (NRC) Management Directive 6.3, "Rulemaking," calls for the consideration of a performance-based alternatives. Such alternatives differ significantly from a prescriptive approach by providing a focus on measurable performance attributes and outcomes. Such alternatives have the potential to improve the objectivity and transparency of NRC decision making and provide greater flexibility to designers and licensees, reduce licensee burden, and promote safety by focusing on safety-successful outcomes.

The process in Management Directive 6.3 is set up in terms of five steps which are intended to provide the information to formulate a performance based alternative to a prescriptive requirement. The five steps in the process are:

1. Defining the regulatory issue and its context
2. Identifying the safety functions
3. Identifying safety margins
4. Selecting performance parameters and criteria
5. Formulating a performance-based alternative.

This document provides useful guidance in developing the risk-informed and performance based process for selecting LBEs that is described in Section 3 of this paper. The use of TLRC in this paper is an example of PB practice at the plant level. Additional consideration of PB practices will also be included in companion LMP papers on RIPB SSC classification and risk informed applications of PRA beyond LBE selection.

A.2 International Guidance for Licensing Basis Event Selection

A.2.1 IAEA Safety of Nuclear Power Plants: Design NSR-1[38]

The International Atomic Energy has published a number of reports on the safety of nuclear power plants including NSR-1 on the design aspects of nuclear safety. This report sets forth basic safety principles to be considered in design to ensure defense-in-depth and high level objectives of the deterministic and probabilistic safety analyses that should be performed to demonstrate adequate safety. The deterministic safety analysis starts with the identification of "Postulated Initiating Events (PIEs)" defined as an event identified during design as capable of leading to anticipated operational occurrences or accident conditions. The following excerpt shows the high level and qualitative character of the guidance for identifying PIEs:

> *A full range of events needs to be postulated in order to ensure that all credible events with potential for serious consequences and significant probability have been anticipated and can be withstood by the design of the plant. There are no firm criteria to govern the selection of PIEs; rather the process is a combination of iteration between the design and analysis, engineering judgement and experience from previous plant design and operation. Exclusion of a specific event sequence needs to be justified.*

Although this document provides useful insights for the design of a nuclear power plant, it was specifically developed for currently operating LWRs and does not appear to provide additional guidance for non-LWRs beyond that which can be similarly gleaned from the US NRC SRP.

A.2.2 United Kingdom Safety Assessment Principles for Nuclear Facilities [39]

The Office for Nuclear Regulation (ONR) is the independent regulator of nuclear safety and security across the United Kingdom. ONR's inspectors use the Safety Assessment Principles (SAPs) described in that document, together with supporting Technical Assessment Guides (TAGs), to guide their regulatory judgements and recommendations when undertaking technical assessments of nuclear site licensees' safety submissions. Supporting these is the legal duty on licensees to reduce risks so far as is reasonably practicable (ALARP) principle informs the use of these SAPs. In addition, the SAPs are used to guide our assessments of proposed new nuclear facilities designs that may come forward for eventual construction at sites in the UK.

What is noteworthy in this document is that the SAPs have already been demonstrated to be technology inclusive and more performance-based that NRC's framework. Earlier versions of these SAPs were used to license two different types of graphite moderated gas-cooled reactors (MAGNOX, and AGR) and a current generation Westinghouse PWR at Sizewell. The 2014 version of the SAPs reflects revisions to account for lessons from the Fukushima Daiichi accident. There are currently SAP-based licensing reviews underway at various stages for new, diverse reactor facilities that may be added to existing sites covering the following designs: a two unit GE-Hitachi ABWR plant, a two unit AREVA EPR plant, a three unit Westinghouse AP1000 plant, and a two unit CGN HPR-1000 plant. In addition there are pre-licensing generic discussions underway for an Integral Pressurized Water Reactor (iPWR)-based SMR designs and a PRISM liquid metal fast reactor.

The SAPs provide useful guidance for the selection of LBEs in this project because they contain the following elements:

- Inclusion of a risk-informed blend of deterministic safety analysis, probabilistic safety analysis, and severe accident analysis
- Numerical criteria for selection of design basis accidents (DBAs) based on frequency of occurrence and resulting from a fault analysis that includes probabilistic and deterministic inputs. Internally initiated accidents less than 10^{-5} per year and accidents initiated by external hazards conservatively estimated to be less than 10^{-4} per year may be excluded from DBAs subject to assurance that risk targets (See BSO and BSL below) and certain deterministic requirements are met.
- Requirements for using a PSA[3] to balance the risk contributions across the design so as to ensure there are no “weak links” in the design that dominate the risks.
- Numerical frequency-consequence criteria for evaluating the acceptability of risks. These criteria are framed in terms of Basic Safety Objectives (BSOs) and Basic Safety Limits (BSLs). Risks below the BSOs are regarded as “broadly acceptable”, those between the BSOs and BSLs “tolerable”, and those exceeding the BSLs “unacceptable. There are BSOs and BSLs for normal operation, design basis accidents, and beyond design basis accidents. Depending on the specific target the consequence limits are applied to workers on-site as well as off-site public exposures.
- In addition to the BSOs and BSLs the SAPs include a requirement to maintain risk levels to as low as reasonably practicable (ALARP) which means that if there are cost effective ways to reduce risk, those means should be implemented regardless of the risk levels relative to BSOs and BSLs.
- When the SAPs are applied to a Generic Design Assessment, which roughly corresponds to a U.S. Design Certification in scope, the frequency basis of the BSO and BSL numerical targets are applied on a reactor-year basis for a generic site. Then when the facility is added to a site, there are separate frequency-consequence criteria applied on an integrated site basis. In this part of the site the equivalent of an integrated site wide assessment of risk is required.

The UK SAPs were found to be very useful in defining the approach to selecting LBEs described in Section 3 including the use of frequency and dose criteria to evaluate the risk of LBEs. This work was also found to be useful in develop the PRA approach that is discussed more fully in a companion white paper. More information on the numerical risk targets employed within the SAPs is provided in Section A.4 below. The SAP are also informative with respect to performance-based criteria development generally. A RIPB SSC classification LMP paper will discuss this further.

[3] IAEA and SAP documents use the term Probabilistic Safety Analysis (PSA) to mean the same thing as Probabilistic Risk Analysis (PRA) according to the IAEA safety term glossary in Reference [38].

A.3 Historical Precedents for Advanced non-LWRs

A.3.1 Modular High-Temperature Gas-Cooled Reactor Pre-Application Review

In the late 1980s and early 1990s, the NRC conducted a pre-application review of the modular high-temperature gas-cooled reactor (MHTGR) at the request of the Department of Energy (DOE). The MHTGR is a graphite moderated, prismatic fueled, helium cooled, high temperature reactor with passive heat removal and reactivity control features.

DOE proposed a systematic, structured method for selecting LBEs that used a top-down approach based on top-level regulatory criteria and PRA. This approach was subsequently built upon in the Exelon, PBMR and DOE NGNP programs, but the MHTGR represents the fullest demonstration of the approach with an actual design, a design specific PRA, and an extensive NRC pre-application to the point of a draft SER, discussed further below.

The scope of the NRC and NRC contractor review for the MHTGR included:

- A Preliminary Safety Information Document (PSID) [40] that included a design description that roughly corresponds to the Safety Analysis Report format in RG 1.70.
- An MHTGR design specific PRA [41] that included:
    - MHTGR-specific initiating events, event sequences, and end states
    - Fault tree models and data to estimate event sequence frequencies
    - Plant transient response analysis for each event sequence
    - Offsite dose consequences for each MHTGR-specific release category.
- A Risk-informed licensing approach based on:
    - Then current LWR requirements and the NRC safety goals
    - Top Level Regulatory Criteria derived from NRC regulations interpreted in the form of a frequency-consequence curve for evaluating the risk significance of LBEs
    - A set of MHTGR design specific LBEs[4] derived from the MHTGR PRA based on probabilistic and deterministic criteria, including AOOs, DBEs, BDBEs, and DBAs.
    - A method for selecting safety-related SSCs based on probabilistic and deterministic criteria and application of the method to the MHTGR
    - Regulatory design criteria for safety-related SSCs during MHTGR-specific LBEs in the performance of MHTGR-specific safety functions.
- Probabilistic and Deterministic Safety analyses for all AOOs, DBEs, selected BDBEs, and DBAs.

The NRC published preliminary results of its MHTGR review in NUREG-1338, "Draft Pre-application Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor." [42] In Sections 3.1.2 of NUREG-1338, the NRC stated that:

> *The staff concludes that the DOE's approach is a systematic and useful approach for design of a nuclear plant. However, it is not an adequate replacement for the application of NRC's regulatory approach to the safety and licensing review. Specifically, the staff found, as a result of review of the MHTGR, that many regulatory criteria (10 CFR) and much Standard Review Plan (NRC report NUREG-0800) guidance are applicable to the MHTGR, and the application of these criteria is necessary to ensure that the MHTGR achieves at least an equivalent level of safety as that of current-generation LWRs.*

In the course of this review, the NRC proposed some additional deterministically selected LBEs be added to the safety evaluation. In response to this, these events were added to the evaluation with an estimate of the frequency of occurrence and an evaluation that no cliff edge effects were identified that would exceed

[4] In the MHTGR submittals, BDBEs were referred to as "Emergency Planning Basis Events (EPBEs), and Design Basis Accidents were referred to a "Safety Related Design Conditions (SRDCs).

the TLRC. No risk significant sequences were identified from these added events. In addition, the NRC review commented that a small number of SSCs that were classified as non-safety related should be added to the safety related list. However the vast majority of the SSC safety classifications were accepted.

A.3.2 Exelon PBMR Pre-Application Review

In 2001 to 2002, the NRC staff conducted a pre-application review of the PBMR design at the request of Exelon. As part of the pre-application engagement with NRC, Exelon proposed a RIPB process similar to the MHTGR process. MHTGR examples were used as well, similar to the LMP examples. In a project closeout letter[5] to Exelon dated March 26, 2002, the NRC staff provided its initial assessment, amongst other things, of the licensing approach proposed by Exelon, including the use of TLRC [48]. With respect to selection of TLRC, the NRC staff stated:

> *"The staff notes that plotting of TLRC is useful to illustrate bounding criteria and safety margins. However, the licensing basis is the set of requirements that are applied to the safety-related equipment to meet the LBEs (or other special regulatory objectives such as anticipated transients without scram (ATWS) or station black-out (SBO)); simply falling within the plot of the TLRC does not in itself constitute a complete licensing basis. Moreover, while the PRA confirms risk insights for a design, and can be used for other purposes as noted above, licensing activities will be a mix of "deterministic" analysis supplemented with risk insights. The lack of operational data for some of the unique PBMR SSCs makes complete reliance on PRA difficult"*

With respect to selection of LBEs, the NRC staff stated in its letter:

> *"In the SRM for SECY-93-092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements," issued April 8, 1993, regarding accident selection and evaluation, the Commission approved the staff recommendation that events and sequences be selected deterministically and use conservative assumptions, and be supplemented with insights from the PRA for the specific design. In Exelon's August 31, 2001, document containing its proposed licensing approach, Exelon appears to be using probabilistic criteria to select AOOs, DBEs, and EPBEs. However, from verbal interactions with Exelon, the staff believes that the candidate LBEs which will be considered for application within the framework of the TLRC will first be established deterministically, and will then be assessed and compared to the TLRC using risk insights. To the extent Exelon adheres to such an approach, the staff believes it would be consistent with previous Commission guidance."*

Following the Exelon review, the NRC staff provided the Commission a status report on the policy implications from licensing non-LWR designs and the staff's plans for seeking Commission guidance on resolving the issues. Three overarching policy issues and four policy issues of a more specific nature were discussed in SECY-02-0139. Of the seven issues, Issue 4, "To what extent should a probabilistic approach be used to establish the plant licensing basis?" specifically relates to LBE selection. The Commission approved the staff's recommendation in the "Staff Requirements Memorandum on SECY-03-0047" to allow the use of a probabilistic approach in the identification of events (See Section A.1.2.1 of this white paper for additional details).

The NRC findings in these reviews for the DOE MHTGR and the Exelon PBMR licensing approaches have been considered in the approach that is described in Section 3. For the LMP project, as was the case with the MHTGR, there is no intent to limit the licensing basis to just meeting the frequency-dose criteria

---

[5] Exelon terminated the PBMR pre-application program as part of a corporate restructuring effort to refocus on core businesses. As a part equity owner in PBMR Pty LTD, this was one of many non-core businesses sold or stopped.

derived from the TLRC. As with currently licensed LWRs, the RIPB approach includes deterministic design criteria for barriers and SSCs that are necessary and sufficient to justify the assumptions made in the PRA on the capabilities and reliabilities of SSCs in the prevention and mitigation of accidents. The collection of papers to be presented on the TI-RIPB approach, including this paper and others on SSC safety classification and defense-in-depth will clarify this key point.

Following the closure of the Exelon PBMR pre-application program, PBMR Pty LTD (PBMR) initiated a pre-application program in early 2004 in its own name. Four RIPB white papers were submitted: PRA; LBE Selection; SSC Classification; and, Defense-In-Depth. This paper on LBE selection also draws from the review of those white papers. In its letter of September 24, 2007, the NRC sent Requests for Additional Information (RAIs) on white papers that had been submitted by PBMR to the NRC for review, including the LBE Selection white paper [45]. Responses to these RAIs were provided by PBMR on March 21, 2008 [46]. Subsequent to the provision of responses to the RAIs, the PBMR licensing project activities were discontinued in May 2010, and the RAI responses were not reviewed by the NRC. These paper were the forerunners to more recent NGNP papers submitted to NRC as part of the NGNP Licensing Strategy.

A.3.3 Next Generation Nuclear Plant (NGNP) Licensing Approach Review

In 2005, the U.S. Department of Energy (DOE) established the Next Generation Nuclear Plant (NGNP) Project at the Idaho National Laboratory (INL) following authorization in the EPACT of 2005. This action supported commercial deployment of a high temperature, gas-cooled reactor (HTGR) technology demonstration plant.

The NGNP project included development of a regulatory framework supportive of commercial HTGR deployment. Framework activities were closely coordinated with the NRC staff and focused on adapting existing nuclear power plant regulatory requirements to the needs of NGNP licensing. The approach for this licensing structure was jointly formulated by DOE and NRC and communicated to Congress in 2008.

NGNP examined HTGR licensing precedents and NRC regulations as they relate to the NGNP safety case and associated plant design goals. The scope and results of this examination were coordinated with and reviewed by NRC staff. In 2009, NGNP used this information to develop a strategic implementation plan for establishing the regulatory basis necessary to complete and submit a HTGR license application to NRC. The plan included:

- *Developing the basis for establishing a mechanistic radiological source term (based primarily on particle fuel design and available qualification testing results)*
- *Preventing/mitigating the release of the radiological source terms to the environment, including methods for the structured and comprehensive identification of licensing basis event sequences, along with establishing multiple radionuclide release barriers*
- *Developing an updated emergency planning structure that considers collocated industry energy end-users to assure protection of public health and safety in the unlikely event of a radiological release.*

A key NGNP methodology in addressing this strategy was to document proposed approaches in a series of complementary pre-licensing “white papers”. Each white paper included a specific set of outcome objectives that support NGNP licensing and was developed with inputs from DOE and the NGNP Licensing Working Group. One of these white papers addressed an approach to selecting licensing basis events (LBEs) [1] and a related paper on an approach to performing a PRA that provided input to the LBE approach [62].[6]

In early 2012, NGNP's DOE/INL team and NRC staff jointly identified and agreed to focus on four key licensing framework topics covering sources of significant regulatory uncertainty for the entire HTGR industry [49]. These topics included:

- HTGR containment functional performance
- Licensing basis event selection
- Mechanistic source terms
- Emergency planning.

Ensuing interactions resulted in NRC staff drafting initial regulatory positions on the four framework topics and submitted them to the NRC's Advisory Committee on Reactor Safeguards (ACRS) for review in early 2013 [50]. Staff findings were then updated and again released in July 2014.

Major items addressed in that NRC staff position report included the following statements relevant to the proposed approach for a risk-informed selection of LBEs:

> *'The licensing basis event identification and categorization process proposed by NGNP included a frequency versus consequence approach for evaluating postulated event sequences against top level regulatory criteria (primarily offsite dose). Initially, based on public meeting discussions and a draft feedback summary written by NRC staff, this approach appeared to be generally reasonable. However, some members of the staff believed that a supplement was probably necessary to DOE/INL's proposed set of design basis accidents (DBAs). This supplement entailed additional deterministically postulated accidents. NGNP personnel felt that adding events from outside the proposed event selection process created significant uncertainty for the industry. The concept of a supplement was also subject to challenge by ACRS recommendations. This issue (and other related topics) was not addressed in the July 2014 NRC staff position report. The omission on this topic, as well as the overall licensing basis event identification and categorization process in general, was attributed to staff concerns that issuing feedback on the topic now might be inconsistent with ongoing NRC efforts related to post-Fukushima Near Term Task Force (NTTF) Recommendation 1 and subsequent development of a risk management regulatory framework. The proposed mechanistic methodology for defining and evaluating source terms was reasonable to NRC staff.*
>
> *The staff was receptive to future emergency planning proposals for a probabilistic risk assessment (PRA) informed approach in sizing the emergency planning zone. Proposals might include use of accident dose assessments when determining an appropriate emergency planning zone size (see NRC's SECY 11-0152 [51], which contains a partial response to NGNP white paper proposals). However, clarification beyond SECY 11-0152 was not provided due to the need for Commission action on related policy issues. Further staff evaluation of the NGNP emergency planning approach was curtailed pending availability of more site and plant design information."*

The Advisory Committee on Reactor Safeguards (ACRS) met May 9-10, 2013 to review the NRC staff's assessment of the NGNP key licensing issues identified in the above paragraphs. The ACRS Subcommittee on Future Plant Designs reviewed INL key licensing issue white papers on January 17,

---

[6] As part of the NGNP Licensing Plan, a longer list of white papers, technical and topical reports were intended to be developed on a timeline consistent with a more complete design and licensing application development program. That program did not materialize and the NGNP – NRC refocused their efforts on a select subset of papers to be submitted to NRC.

2013, and staff assessments of the INL white papers on April 9, 2013. The following conclusions and recommendations were reached by the ACRS [50]:

1. *"The staff assessment of the NGNP white papers on key technical issues is appropriate, given the unavailability of many plant-specific design details, such as the selected fuel form (pebble or prismatic) and a complete plant design. The final assessments should be published after the issues raised in Recommendations 2, 3, and 4 are addressed.*
2. *The assessment documents should be revised to provide clear links to the numerous requests for additional information (RAIs) and responses that were developed during their assessment because the white papers have not been revised to incorporate those agreements.*
3. *The licensing basis event selection assessment should point out the need to clarify the definition of event sequences and event sequence families to ensure consistency in developing licensing basis events and design basis accidents (DBAs). Incoherent logic in the event trees should be addressed.*
4. *The staff's suggestion that the final selection of DBAs include postulated deterministic event sequences is inconsistent with a risk-informed framework proposed by the NGNP project and with other on-going NRC activities encouraged by the Commission. Although engineering judgment may be invoked to include postulated deterministic event sequences in the final selection of DBAs, if such sequences are not in the probabilistic risk assessment (PRA), the PRA is incomplete and should be revised to include them. They then can be fully evaluated and considered for inclusion as DBAs."*

The approach to selection of LBEs described in Section 3 of this white paper benefitted from the guidance offered in the NRC and ACRS reviews of the NGNP white papers summarized in the previous paragraphs.

A.3.4 PRISM Pre-Application Review

Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor is part of DOE's advanced liquid-metal reactor program. PRISM is a small, modular, pool-type, liquid-metal (sodium)-cooled reactor. Multiple modules are expected to constitute a power block and multiple power blocks may be combined to constitute a power station of more than 1000 MWe. Each reactor module would be a standard design that would be built in a factory and shipped by rail to a site. PRISM uses an advanced metal fuel (plutonium-uranium-zirconium) alloy inside cladding of ferritic steel alloy (HT9) within a fuel rod assembly arrangement. The core structure material is also HT9. Six control rods provide the necessary operational reactivity control. The standard plant design for the PRISM consists of three identical power blocks with a total electrical output rating of 1395 MWe. Each power block comprises three reactor modules, each with an individual thermal rating of 471 MWt. Each reactor module is located in its own below-grade silo and is connected to its own intermediate heat transport system and steam generator system. The design includes passive reactor shutdown and passive decay heat removal features.

PRA is employed as a design tool for PRISM. A preliminary PRA was used in the conceptual design phase to define a set of accident sequences from initiating event to radiological release into the environment. PRA is treated as an essential part of the design process providing essential safety inputs to the design.

The PRISM PRA was used to help ensure completeness in the identification of accident sequences and to rank the sequences in order of their importance based on their expected occurrence frequency and offsite consequences. Licensing basis events were categorized as either Design Basis Accidents or Beyond Design Basis Accidents using the event sequences. All event sequences were considered as candidate LBE's. Specific guiding criteria include:

1. Event sequences with frequency greater than 10^{-6} per reactor year are within the design basis event envelope. There are four categories of DBEs including “normal operation” including events with frequencies greater than 10^{-1} per reactor year, “anticipated events” with frequencies between 10^{-1} and 10^{-2} per reactor year, “unlikely events” with frequencies between 10^{-2} and 10^{-4} per reactor year, and “extremely unlikely events” with frequencies between 10^{-4} and 10^{-6} per reactor-year. Events with frequencies below 10^{-6} per reactor-year were classified as “beyond design basis events”.
2. Within the design basis event envelope, events of greater severity shall have lower frequency.

NRCs Pre-application Safety Evaluation Report (PSER) of PRISM Liquid-Metal Reactor and the supporting PRA is documented in Reference [43] completed in 1994. The PRISM conceptual design was submitted by the U.S. Department of Energy (DOE) in accordance with the NRC's "Statement of Policy for the Regulation of Advanced Nuclear Power Plants" (51 Federal Register 24643). This policy provides for the early Commission review and interaction with designers and licensees. The PRISM reactor design proposed by DOE is for a small, modular, pool-type, liquid-metal (sodium)-cooled reactor.

The approach followed in the selection of LBEs to support this review is captured in the following excerpt from the PSER:

> *The methodology used by General Electric (GE) for defining the design-basis events (DBEs) for the PRISM reactor is described in Chapter 15 of the PSID. The procedure is systematic and draws upon PRA work performed in the conceptual stage of the design. The PRA is used to help ensure completeness in the identification of accident sequences and to rank the sequences in order of their importance on the basis of their expected occurrence frequency and offsite consequences. Each event is placed into a category of either a DBE or a beyond-design-basis event (BDBE). GE has considered all events occurring at a frequency of 10^{-6} or more per reactor-year to be DBEs. GE analyzes these events in a conservative manner. Less likely events are considered BDBEs (frequencies < 10^{-6} per reactor-year). GE considers these off-normal conditions of such extremely low probability that no event in this category is considered credible during the plant's lifetime. BDBEs can, however, have significant consequences. GE acknowledges some of these events may merit consideration in establishing the design. These BDBEs are discussed in Appendices E and G of the PSID.*

In its review, the NRC staff used a somewhat different definition of design basis event categories than proposed by GE based on expected frequency of occurrence.

In addition to reviewing the results of the PRISM PRA for the purpose of event identification, the NRC staff defined a set of postulated “Bounding Events” that were added to Category EC-III to support the safety evaluation. The methodology used to define these bounding events was not provided so it appears to be based on the staff’s engineering judgment. Subjective estimates of the frequency of occurrence of these bounding events were provided which included estimates as low as 10^{-10} per reactor year, including one extreme event involving removal of all control rods with failure to scram, and station blackout events lasting as long as 36 hours. There are no comparably extreme events considered in licensing currently operating LWR plants.

The PRISM PRA that was reviewed was performed for a single reactor module so the question of multi-module accidents did not come up in the NRC review. However, more recently GE-Hitachi has completed an upgrade to the PRISM PRA to pilot the ASME/ANS PRA Standard for advanced non-

LWRs [32] and multi-module accidents were addressed in the more recent work. More information on the PRISM PRA is provided in a companion white paper on PRA development to support this LMP.

A.3.5 ANS Design Standard for Modular Helium Cooled Reactor Plants [25]

This standard, ANS 53.1 “Modular Helium Reactor Safety Design” was issued in 2011 to provide a risk-informed and performance-based design process for modular helium cooled reactors (MHRs). The purpose of this standard is to provide nuclear safety criteria applicable to the design of MHRs leading up to the preparation of a license. To achieve this purpose, this standard provides a process that can be used to:

- develop MHR top-level nuclear regulatory safety criteria;
- identify safety functions, top-level design criteria, licensing-basis events, design basis accidents, and methods for performing safety analyses;
- determine safety classification of systems, structures, and components (SSCs)
- identify safety-related SSC special treatment requirements and defense-in-depth (DID) provisions;
- demonstrate the adequacy of DID by applying a risk-informed evaluation approach.

The standard was influenced by the RIPB design and licensing approach employed for the MHTGR and subsequently refined in the Exelon, PBMR and NGNP white papers. The flow chart used to describe the process advanced in this standard shown in Figure 2 is essentially the same as a flow chart found in the NGNP Defense-in-Depth Approach white paper [26]. An important contribution made in this standard is specific guidance for the designer to implement the processes described at a high level in the NGNP white papers on LBE selection, PRA development, SSC safety classification, mechanistic source term development, and evaluation of defense-in-depth (DID) adequacy. With influence from that same white paper, this standard included specific design criteria for evaluating the adequacy of defense-in-depth. These criteria are derived from the DID principles listed in **Table** 3 and provide examples of DID principles to evaluate the selection of LBEs as discussed more fully in Section 3.

A.3.6 Yucca Mountain Pre-closure Safety Analysis (PCSA) [27]

Additional guidance for selecting LBEs was found in the case of the Yucca Mountain Pre-Closure Safety analysis which was submitted and successfully reviewed by the NRC staff [28]. It is relevant to this project because it is an example of a safety and regulatory evaluation of a first-of-a-kind nuclear facility, it utilized a risk-informed process to inform the design and to select LBEs, made use of frequency and dose criteria tied to the regulations, and exhibits key elements of a technically sound and well structured, risk-informed and performance-based design making process.

Congress established the bases for NRC licensing of a geologic high level nuclear waste (HLW) repository at Yucca Mountain. NRC was authorized to exercise its licensing and regulatory authority under the Atomic Energy Act of 1954, as amended (42 U.S.C. 2011 et seq.), to license and regulate a DOE facility for the disposal of HLW, including spent nuclear fuel (SNF). Subsequently, Congress authorized the NRC to promulgate technical requirements and criteria that it would apply to determine whether to approve or disapprove of DOE applications to construct a repository, receive and possess HLW and SNF in a repository, and close and decommission a repository.

Table 3 Defense-in-Depth Principles from ANS 53.1

1	Radionuclide release barriers are sufficiently robust to withstand challenges identified for the design.
2	Each barrier's failure probability is acceptably low compared with identified challenges.
3	As-designed, built, and maintained multiple radionuclide release barriers minimize dependencies. Events that challenge two or more barriers are infrequent, and the postulated failure of one barrier does not significantly increase the failure probability of another barrier.
4	Overall barrier redundancy and diversity ensure compatibility with the TLSC.
5	Accidents potentially releasing significant radioactive material quantities preserve a reasonable prevention/mitigation balance.
6	Safety design avoids overreliance on programs to compensate for plant design weaknesses.
7	System redundancy, independence, and diversity cover expected challenges based on frequency, system failure consequences, and associated uncertainties.
8	The safety design adequately addresses common-cause failures.
9	Performance of a risk-significant safety function is not reliant on a single engineered feature except where inherent safety is demonstrated for all failure modes.
10	The approach evaluates human-error likelihood and consequences, thus providing defenses against human errors that can lead to significant radioactive material release.
11	The design meets the GDC intent applicable in 10 CFR 50, Appendix A, and reactor-specific regulatory design criteria from RI-PB licensing.

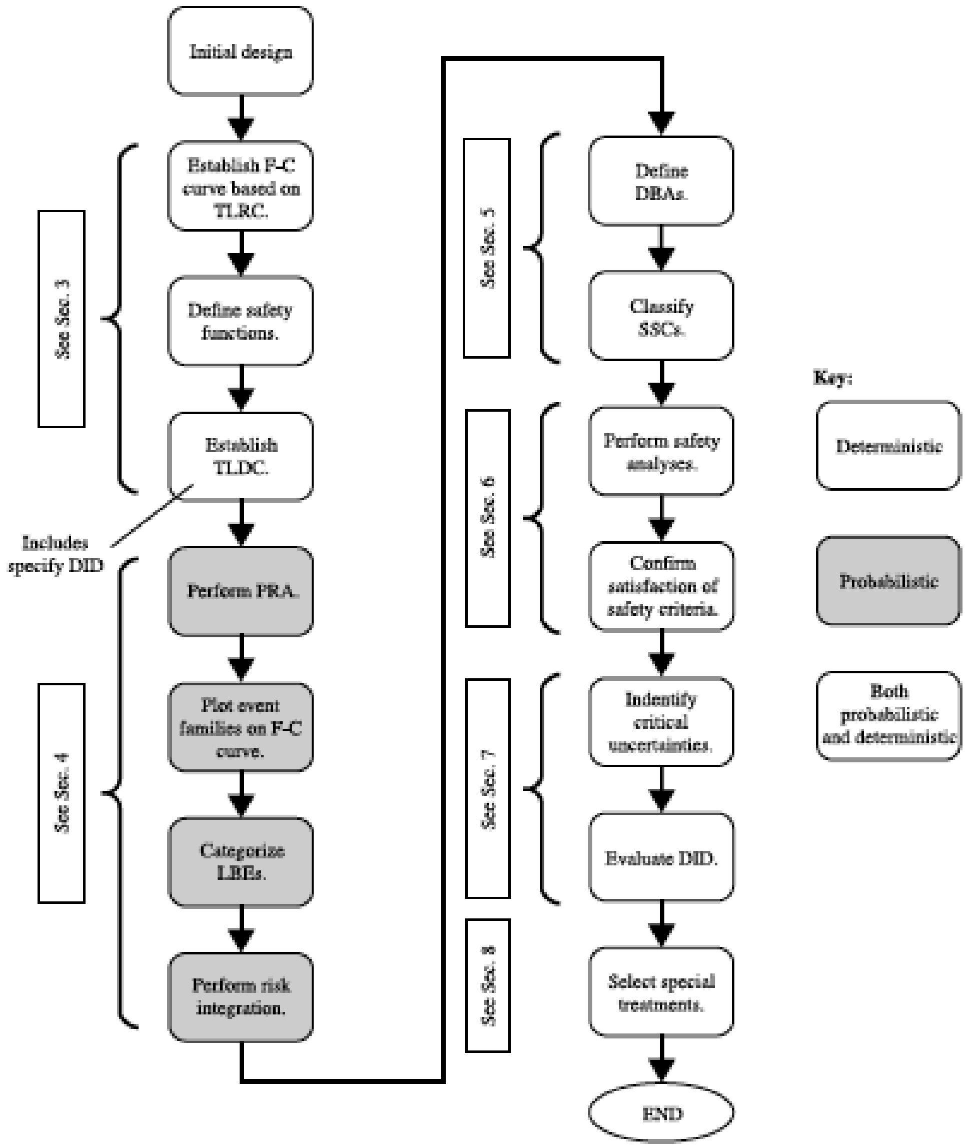

Figure 2 MHR Safety Design Process in ANS 53.1

In 2001, the NRC issued its technical requirements and criteria in 10 CFR Part 63 [29]. In 2003, the NRC issued NUREG-1804, Rev 2, The Yucca Mountain Review Plan [30]. This plan provides guidance for the NRC staff to evaluate a U.S. Department of Energy license application for a geologic repository. Unlike licensing associated with 10 CFR Part 50, regulatory guides were not associated with 10 CFR Part 63.

The regulation and the review plan address the pre-closure period of operation in which waste is emplaced into Yucca Mountain and the period thereafter in which the permanent closure and permanent storage within Yucca Mountain occurs. DOE is the licensee and established nominally 100 years as the preclosure period for purposes of safety analysis. This section discusses only the safety analysis and interaction with the design and engineering associated with the pre-closure period.

According to the 10 CFR Part 63.2 definition, an initiating event means a natural or human-induced event that causes an event sequence. Consistent with this definition, an initiating event is a departure from normal operation that triggers an event sequence. These definitions are consistent with those used in nuclear reactor PRAs [31]. As defined in 10 CFR Part 63.2, event sequence means a series of actions or occurrences or both within the natural and engineered components of a geologic repository operations area (GROA) that could potentially lead to exposure of individuals to radiation.

Combining the 10 CFR Part 63.2, 10 CFR Part 63.111 and 10 CFR Part 63.204 definitions, important to safety (ITS), with reference to structures, systems, and components, means those engineered features of the geologic repository operations area whose function is:

1. To provide reasonable assurance that high-level waste can be received, handled, packaged, stored, emplaced, and retrieved such that no member of the public in the general environment receives more than an annual dose of 0.15 mSv (15 mrem) during normal operation and Category 1 event sequences,
2. To prevent or mitigate each Category 2 event sequence that could result in radiological exposures to any individual at or beyond the site boundary, that could result in the more limiting of a TEDE of 0.05 Sv (5 rem), or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The lens dose equivalent may not exceed 0.15 Sv (15 rem), and the shallow dose equivalent to skin may not exceed 0.5 Sv (50 rem).

Table 4 presents the performance criteria that were developed. Category 1 event sequences are those that could occur at least once over the pre-closure period which is equivalent to a frequency at least 10-2 per year. Category 2 event sequences are those that could occur with frequency less than 10-2 per year but greater than or equal to a frequency of 10-6 per year. Less frequent event sequences are termed “Beyond Category 2”. Performance objectives were not specified for Beyond Category 2 event sequences. There are no NRC equivalents to QHOs defined for such events for Yucca Mountain.

10 CFR Part 63.111 further states that a preclosure safety analysis (PCSA) must be performed and 10 CFR Part 63.112 specifies the scope of that analysis. As the licensee, the DOE elected to use a combination of probabilistic risk assessment methods, hazard analysis methods and deterministic methods as the basis for the PCSA. The analysis demonstrated compliance with the performance objectives for internal events, on-site hazards, off-site hazards, and natural phenomena hazards. DOE submitted a license application under this regulation in June 2008. Requests for Additional Information were made and answered during 2009 and 2010. The NRC’s five volume Safety Evaluation Report was issued over the period 2010 to 2015 with a positive assessment of the PCSA. During the development of the license application submitted in 2008, the design and PCSA proceeded concurrently such that engineering and design were heavily influenced by the PCSA and visa-versa. Figure 3 is an overview of the risk-managed design process emphasizing the pre-closure safety analysis steps.

The PCSA, as is typical of a high quality PRA, may be thought of as a simulation of how a facility with its systems and personnel acts and reacts when something goes wrong (i.e. in response to an initiating event). The PCSA identified design bases and procedural safety controls for ITS SSCs that prevent (i.e., reduce the likelihood of) or mitigate (i.e., reduce the severity of) event sequences. The PCSA also provided inputs for developing license specifications as well as management, maintenance, training, and operations programs that ensure the availability of ITS SSCs. The PCSA was a collaborative effort with repository design groups. Preliminary event sequences were identified early in the design, and safeguards were incorporated into the design to reduce event sequence probabilities, including those that involved human error as well as hardware. As a practical matter, the safety analysis staff reviewed every aspect of the design relevant to the nuclear safety model (e.g., the event sequences and hazard analyses). As more detail was added to the design, more detail was added to the models. When a performance objective was in jeopardy, the safety analysis staff defined what needed to be changed either to mitigate or prevent the event sequence or hazard. It was then the responsibility of the engineering and design staff to modify the design criteria and specifications to achieve the nuclear safety objectives. The PCSA, therefore, was an integral part of the design process.

Table 4 Performance Criteria for Category 1 and 2 Event Sequences and Normal Operation

Event Sequence Type	Category of Individual	GROA[a] Restricted Areas	Site (Preclosure Controlled Area)	Offsite[b] in the General Environment (Unrestricted Area)	Offsite[b], but not within the General Environment (Unrestricted Area)
Aggregate of Normal Operation and Category 1 Event Sequences Dose (Category 1—Those event sequences that are expected to occur one or more times before permanent closure of the GROA)[c]	Public	—	100 mrem/yr[d,e,f,g]	15 mrem/yr[h,i] 2 mrem in any hour[g]	100 mrem/yr[d,e,f] 2 mrem in any hour[j]
	Radiation worker[k,l]	5 rem/yr[d,e,m] 50 rem to any organ 15 rem lens of eye 50 rem skin	See note n.	See note n.	See note n.
Single Category 2 Event Sequence Dose (Category 2—Other event sequences that have at least one chance in 10,000 of occurring before permanent closure of the GROA)[c]	Public	—	—	5 rem[o] 50 rem to any organ 15 rem lens of eye 50 rem skin	5 rem[o] 50 rem to any organ 15 rem lens of eye 50 rem skin

NOTE: [a]Other areas of the site may be identified as restricted areas as required by operations.
[b]Offsite areas are areas outside of the preclosure controlled area (See Figure 1.8-2).
[c]10 CFR 63.2.
[d]10 CFR 63.111(a)(1).
[e]10 CFR 63.111(b)(1).
[f]10 CFR 20.1301(a)(1).
[g]10 CFR 20.1301(a)(2)(b).
[h]10 CFR 63.111(a)(2).
[i]10 CFR 63.204.
[j]10 CFR 20.1301(a)(2).
[k]Individual with assigned duties involving exposure to radiation or to radioactive material.
[l]Occupational doses are those received during the course of those assigned duties.
[m]10 CFR 20.1201.
[n]If receiving an occupational dose (see note k above) at this location, the GROA restricted areas' occupational objectives apply; otherwise, the individual is considered a member of the public.
[o]10 CFR 63.111(b)(2).

Design, site, and operational information from various disciplines were inputs to the PCSA. Design information used to identify the initiating events and to conduct the event sequence analyses was obtained from design documents, such as design drawings, design reports, piping and instrumentation diagrams, control logic diagrams, and design calculations. Design information on locations and amounts of radioactive material present was used in performing consequence and criticality analyses. Site information, such as wind patterns, proximity of potentially hazardous materials, and seismicity, was also used in the PCSA particularly for natural and man-made on-site and off-site hazard analyses. Representative waste containers, rather than those of specific vendors, were analyzed for their failure potential associated with event sequences. A range of radioactive HLW and SNF container dimensions and materials and internal configurations were considered within these representative analyses.

The experience with the PCSA at Yucca Mountain was found to be very useful in defining the approach to selecting LBEs described in Section 3 including the use of frequency and dose criteria to evaluate the risk of LBEs. The fact that a PRA approach was used to meet safety and licensing requirements for a new facility that lacked any relevant deterministic regulatory precedents is significant. In addition the NRC regulations for this new facility included specific numerical criteria for the frequencies and doses of LBEs that were used to establish the original licensing basis. Finally the Yucca Mountain Pre-Closure Facility licensing approach included the development of performance requirements for SSCs in the prevention and mitigation of accidents. This work was also found to be useful in develop the PRA approach for LBE selection, the RIPB approaches to SSC safety classification and DID that are discussed more fully in companion white papers.

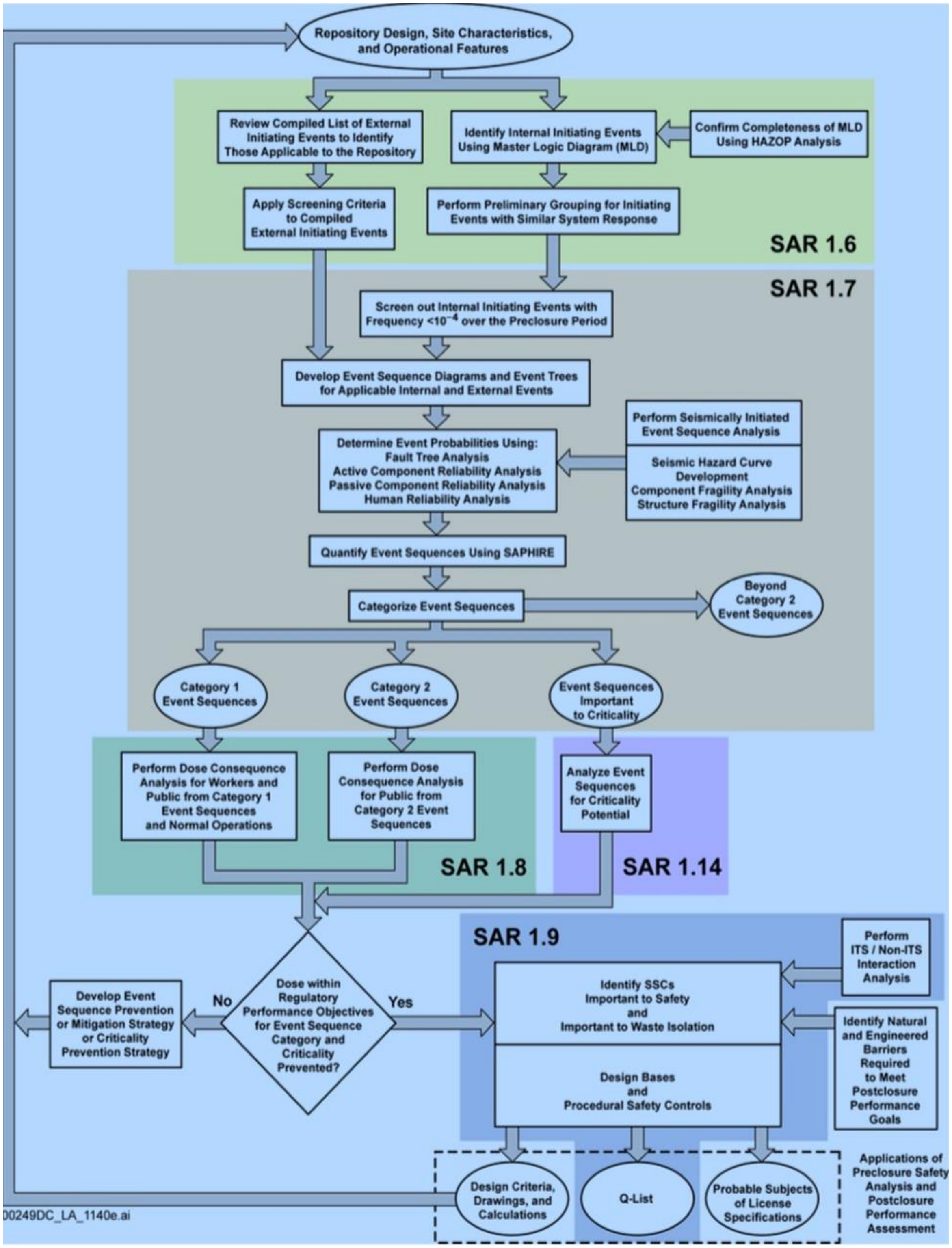

Figure 3 Use of the PCSA for Risk Management of Repository Design

evaluation criteria to evaluate the risk significance of LBEs in terms of their frequencies and consequences. This evaluation is performed in order to assure public safety and to assess the adequacy of the performance of SSCs that perform safety functions during these LBEs. The TLRC frequency-consequence evaluation criteria are based on the following objectives:

1. Provide direct public health and safety acceptability evaluation boundaries in terms of individual radiological consequences, i.e., performance-based criteria
2. Are independent of reactor type and site
3. Provide well-defined, quantifiable risk criteria.

The following primary sources have been identified as containing criteria or concepts that can be used to establish evaluation boundaries on the risk or consequences of potential radiological releases from nuclear power plants in the United States. Each was considered in this paper. Not all are equivalent or individually additive, thus requiring some judgement in synthesizing the TLRC frequency-consequence criteria for the LMP.

- **Reactor Safety Goal Policy Statement [11]:** On August 4, 1986, the NRC adopted a safety goal policy for the operation of nuclear power reactors. The objective of this policy is to establish goals that broadly define an acceptable level of radiological risk. Two qualitative safety goals supported by two Quantitative Health Objectives were established. These two supporting objectives are based on the principle that nuclear risks should not be a significant addition to other societal risks.

  This policy limits public safety risk resulting from nuclear power plant operation. Limits are stated in the form of the maximum allowable risk of immediate death and the risk of delayed mortality from exposure to radiological releases of all types from nuclear power plants.
- **10 CFR Part 20, "Standards for Protection against Radiation (Subpart C, Occupational Dose Limits)"[52]:** The regulations promulgated under 10 CFR Part 20 establish standards for protection against ionizing radiation resulting from activities conducted under licenses issued by the NRC. Event sequences expected to occur within the plant lifetime, considering multiple reactor modules, are classified as AOOs. AOOs are evaluated against the dose limits derived from annual dose limits in 10 CFR Part 20.
- **10 CFR Part 20, "Standards for Protection against Radiation (Subpart D, Radiation Dose Limits for Individual Members of the Public)"[53]**: These criteria (§20.1301) specify annual dose limits for releases associated with relatively high frequency events that occur as part of normal plant operations.
- **10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents"[54]**: This appendix provides explicit annual limits on doses from planned discharges that meet the NRC's definition of ALARA.
- **10 CFR Part 52, Subpart C, "Combined Licenses"[55]:** Under the provisions of 10 CFR Part 52.79 [56], an application for a combined license must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for SSCs important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and

safety of the public. This standard repeats the same dose requirements as specified in 10 CFR Part 100 and 10 CFR Part 50.34(a)(ii)(d).

- **40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations"[57]:** These standards provide the generally applicable exposure limits for members of the general public from all operations except transportation and disposal or storage of spent fuel associated with the generation of electrical power by nuclear power plants.
- **10 CFR Part 100, "Reactor Site Criteria (Subpart B, Evaluation Factors for Stationary Power Reactor Site Applications on or After January 10, 1997)"[58]**: §100.20 defines the EAB and low population zones (LPZs) of a nuclear reactor site, and requires that the combination of the site and reactor located on that site be capable of meeting the dose and dose rate limitations set forth in 10 CFR §50.34(a).
- **10 CFR §50.34(a)(ii)(d), "Contents of Applications: Technical Information "[59]:** This section of the regulation specifies dose limits for evaluating the acceptance of the engineered safety features that are intended to mitigate the radiological consequences of accidents. These dose limits are consistent with those utilized in 10 CFR Part 100 for determining the extent of the EAB and Emergency Planning Zone (EPZ).

NUREG-0800, Standard Review Plan Chapter 15.0 Introduction - Transient and Accident Analyses [65]: This document specifies acceptance criteria for AOOs and states the principle that the risks of AOOs and postulated accidents as defined by the product of the frequency and consequence should be about the same. The acceptance criteria permit the doses from lower frequency AOOs to be greater than the annual dose limits in 10 CFR 20 as long as other acceptance criteria, including the need to avoid restrictions on uncontrolled areas are met.

- **NUREG-1860 Frequency-Dose Criteria [21]:** This document, reviewed in the previous section, proposes an approach for identifying and classifying licensing basis events into frequency categories, and frequency-dose criteria for evaluating the risks of radiological exposures.
- **United Kingdom Safety Assessment Principles [39]:** The UK SAPs, reviewed in the previous section, provide frequency-consequence criteria for evaluation of risks at the reactor level and at the integrated site level.

Each of these primary sources is discussed in greater detail below and their use in the selection of frequency-consequence evaluation criteria in Section 3. The U.S. regulations have been grouped into three sets of criteria, consistent with the category of event(s) to which they apply. The UK SAPs, which span the full spectrum of LBEs are discussed in a separate section.

A.4.1 TLRC Related to Normal Operation and AOOs

10 CFR §50.34, 10 CFR Part 20, and Appendix I of 10 CFR Part 50 all provide guidance on the limits for radiological releases from reactors during normal operations.

The regulations do not define the term 'normal operation' in quantitative terms, i.e., the expected frequency of specified anticipated occurrences. However, Appendix A to 10 CFR Part 50 defines AOOs as "those conditions of normal operation... expected to occur one or more times during the life of a nuclear power plant."

NUREG-0800 SRP Chapter 15.0 specifies acceptance criteria for AOOs and states the principle that the risks of AOOs and postulated accidents as defined by the product of the frequency and consequence should be the same. The acceptance criteria permit the doses from a specific AOO to be greater than the annual dose limits in 10 CFR 20 as long as other acceptance criteria are met. The following quotes from NUREG-0800 Chapter 15.0 elaborate on these points:

> *"If the risk of an event is defined as the product of the event's frequency of occurrence and its consequences, then the design of the plant should be such that all the AOOs and postulated accidents produce about the same level of risk (i.e., the risk is approximately constant across the spectrum of AOOs and postulated accidents). This is reflected in the general design criteria (GDC), which generally prohibit relatively frequent events (AOOs) from resulting in serious consequences, but allow the relatively rare events (postulated accidents) to produce more severe consequences."*

In specifying the acceptance criteria for the least likely AOO events, referred to by the ANS classification "Condition III" events, the following criterion is provided:

> *"For PWRs, the release of radioactive material may exceed guidelines of 10 CFR Part 20, but shall not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius.*
>
> *For BWRs, the offsite release of radioactive material is limited to a small fraction of the guidelines of 10 CFR Part 100, which may be the result of the failure of a small fraction*[7] *of the fuel elements in the reactor."*

The combination of these statements supports the view that the annual dose limits in 10 CFR 20 should be interpreted as annual risk limits and not limits to be applied to individual AOO events of low frequency . For the least likely (Condition III) events in the AOO category, these acceptance criteria clearly suggest that dose per event limits can be proportionally greater as long as the annual risk limit is not exceeded. This insight is used to revisit the derivation of the TLRC frequency-dose criteria that was developed in the NGNP LBE White Paper [1], as discussed more fully in Section 3.

10 CFR §20.1301 requires that the TEDE for a member of the public be limited to 100 mrem "from licensed operation… in a year" and 2 mrem in any 1 hour, in unrestricted areas. Presumably, unrestricted areas means off-site and at the site boundary. This regulation provides the applicable criteria for limiting dose to the general public from anticipated and unanticipated events associated with the normal (nonaccident) operation of a nuclear power plant.

10 CFR Part 50, Appendix I, identifies dose and dose rate limits and limits on planned releases from the operation of nuclear power plant rad-waste systems during normal operation, to maintain exposures ALARA. These criteria provide implementation guidance for applying the requirements of 10 CFR §50.34(a) and §50.36(a), for planned releases from the radwaste systems of nuclear power plants to the general environment to be ALARA. These requirements specify the following limits:

> *"The applicant shall provide reasonable assurance that the following design objectives will be met.*
>
> *A. The calculated annual total quantity of all radioactive material above background to be released from each light-water-cooled nuclear power reactor to unrestricted areas will not result in an estimated annual dose or dose commitment from liquid effluents for any individual in an*

[7] Although the term "small fraction" is not quantified in this part of the SRP, in Chapter 15.0.3 the term "small fraction" of another dose limit is defined as 10% of the referenced dose limit.

unrestricted area from all pathways of exposure in excess of 3 millirems to the total body or 10 millirems to any organ.
B.1. The calculated annual total quantity of all radioactive material above background to be released from each light-water-cooled nuclear power reactor to the atmosphere will not result in an estimated annual air dose from gaseous effluents at any location near ground level which could be occupied by individuals in unrestricted areas in excess of 10 millirads for gamma radiation or 20 millirads for beta radiation."

A.4.2 TLRC Related to DBEs

10 CFR §50.34(a)(1) contains NRC's regulations governing the design of new reactors and the means provided to protect against DBAs. This LBE selection approach uses the term 'events' in lieu of 'accidents' (as found in the regulations discussed below) for the identification of unplanned, off-normal events not expected in the plant lifetime.

10 CFR §50.34(a)(1) requires that any reactor be designed such that:

> *"An individual located at any point on the EAB would not receive a radiation dose in excess of 25 rem TEDE for any 2-hour period following the onset of a postulated fission product release.*
>
> *An individual located at any point on the outer boundary of the LPZ, exposed to the radioactive cloud resulting from a postulated fission product release, would not receive a radiation dose for any 30-day period in excess of 25 rem TEDE."*

10 CFR §50.34(a)(ii)(D) requires that these consequence limits be used when evaluating the acceptability of the features included in the plant design (i.e., engineered safety features and fission product barriers) for mitigating accident radioactive releases. The footnote pertaining to this section states that the fission product release to be assumed should be based "upon a major accident... postulated from consideration of possible accidental events." 10 CFR §100.21(c)(2), "Reactor Site Criteria: non-seismic site criteria," requires that the radiological dose consequences of postulated accidents meet the criteria stated in 10 CFR §50.34(a)(1) for the type of facility located at the site in question.

In general, NRC's regulations do not define the type of events that comprise the category of DBAs. For LWRs, the GDC (Appendix A to 10 CFR Part 50) indicates that LOCAs must be considered as postulated accidents when designing safety systems. However, the Standard Review Plan §15.0 includes a list of generic events appropriate for LWRs some of which may be applicable to other reactor types. It is stated in the SRP that the applicant is expected to define the appropriate "limiting" events for its design and sufficient design information for the staff to review the event selection. However, there is no method specified or referenced for ensuring that the appropriate set of "limiting" design specific events have been identified.

The regulations do not define DBAs in terms of their expected frequencies of occurrence, but 10 CFR §50.34(a)(i)(2) articulates the expectation that the design, construction, and operation of nuclear power reactors will be such as to produce an 'extremely low probability of occurrence' for accidents that could release significant quantities of radioactive fission products. No quantitative definition of the term 'extremely low probability' is provided in the regulation.

In the licensing of advanced non-LWRs, the NRC has introduced review acceptance criteria that clarifies the association of the 10 CFR §50.34(a)(i)(2) dose thresholds with the extremely unlikely large LOCA

DBA, but expects that more likely DBAs be held to more stringent dose thresholds. The following Table 5 lists different events with different dose thresholds [63]:

The explanation for assigning different dose criteria given for this table is as follows:

> *"The dose acceptance criteria in Table 1(Table 5 below) of this SRP section are fractions of the 10 CFR 50.34(a)(1) dose reference values for accidents other than the LOCA, as has been done historically. For events having a moderate frequency of occurrence, any release of radioactive material must be such that the calculated offsite doses are a small fraction of the 10 CFR 50.34(a)(1) reference values. A small fraction is defined as less than 10% of the 10 CFR 50.34(a)(1) reference values, or 2.5 rem TEDE. The plant site and dose mitigating engineered safety features are acceptable with respect to the radiological consequences of a postulated control rod drop accident (BWR), control rod ejection accident (PWR), fuel handling accident or cask drop accident if the calculated offsite doses are well within the dose reference values in 10 CFR 50.34(a)(1). "Well within" is defined as 25% of the 10 CFR 50.34(a)(1) reference values, or 6.3 rem TEDE."*

The assumptions made by the staff in associating events with either a 25% dose threshold or 10% dose threshold to a numerical frequency of occurrence are not specified. A search of the supplied references was not successful in establishing the staff's estimates of the event frequencies or other basis for using these more limiting dose thresholds for DBAs more likely than the limiting event. As a result, this information is not directly used in developing frequency-consequence criteria. However the idea that the acceptable doses should be linked to the frequency of occurrence of any event is utilized in selecting the LMP frequency-consequence evaluation criteria.

Table 5 Accident Dose Criteria (Table 1 from Reference [63])

Accident or Case	EAB and LPZ Dose Criteria	Analysis Release Duration
LOCA	25 rem TEDE	30 days for all leakage pathways
BWR Main Steam Line Break		Instantaneous puff, until MSIV isolation
Fuel Damage or Pre-incident Spike	25 rem TEDE	
Equilibrium Iodine Activity	2.5 rem TEDE	
BWR Rod Drop Accident	6.3 rem TEDE	24 hours
Small Line Break Accident	2.5 rem TEDE	Until isolation, if capable, or until cold shutdown is established
PWR Steam Generator Tube Rupture		Affected SG: time to isolate; Unaffected SG(s): until cold shutdown is established
Fuel Damage or Pre-incident Spike	25 rem TEDE	
Coincident Iodine Spike	2.5 rem TEDE	
PWR Main Steam Line Break		Until cold shutdown is established
Fuel Damage or Pre-incident Spike	25 rem TEDE	
Coincident Iodine Spike	2.5 rem TEDE	
PWR Locked Rotor Accident	2.5 rem TEDE	Until cold shutdown is established
PWR Rod Ejection Accident	6.3 rem TEDE	30 days for containment leakage pathway; Until cold shutdown is established for secondary pathway
Fuel Handling Accident or Cask Drop	6.3 rem TEDE	2 hours

A.4.3 TLRC Related to Policy Guidance for BDBEs

Current policy and guidance require that certain events outside the scope of the normal operation and DBE categories be considered in the design of nuclear power plants.

The NRC's "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants" states the Commission's intent to "…take all reasonable steps to reduce the chances of occurrence of a severe accident involving substantial damage to the reactor core and to mitigate the consequences of such an accident should one occur." As noted earlier, this policy statement specifically addresses the Commission's intent to resolve safety issues associated with "accidents more severe than design basis accidents." This policy statement provides the following criteria for evaluating new designs for safety adequacy for addressing severe accident potential.

> *The Commission believes that a new design for a nuclear power plant (as well as a proposed custom plant) can be shown to be acceptable for severe accident concerns if it meets the following criteria and procedural requirements:*
>
> a. *Demonstration of compliance with the procedural requirements and criteria of the current Commission regulations, including the Three Mile Island requirements for new plants as reflected in the CP Rule (10 CFR 50.34(f)):*
> b. *Demonstration of technical resolution of applicable unresolved Safety Issues, including a special focus on assuring the reliability of decay heat removal systems and the reliability of both AC and DC electrical supply systems:*
> c. *Completion of a Probabilistic Risk Assessment (PRA) and consideration of the severe accident vulnerabilities the PRA exposes along with the insights that it may add to the assurance of no undue risk to public health and safety:*
> d. *Completion of a staff review of the design with a conclusion of safety acceptability using an approach that stresses deterministic engineering analysis and judgment complemented by PRA.*

In addition to its Severe Accident Policy, the Commission has issued NUREG-0880, "Safety Goals for Nuclear Power Plant Operation" and the related policy statement entitled "Safety Goals for the Operation of Nuclear Power Plants." [11] Two qualitative safety goals are used to express the Commission's policy regarding the acceptable level of radiological risk from nuclear power plant operation as follows:

- *Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.*
- *Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.*

The following Quantitative Health Objectives (QHOs) were identified as the basis for determining achievement of the above safety goals:

> *The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.*

The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

The statement of risks provided in the Safety Goal Policy envelops the spectrum of allowable risk associated with the operation of a nuclear power plant. As such, it clearly defines the outermost boundaries of acceptable risk associated with any event that has the potential to produce a radiological release affecting the environment or the health and safety of the general public.

A.4.4 Criteria for Classifying LBEs Based on Frequency of Occurrence

In its June 26, 1990, SRM on SECY-90-16 [35], the Commission endorsed a core damage frequency (CDF) goal of 10^{-4} per year for advanced reactors. Since accidents involving severe core damage are considered beyond the design basis this implies that DBAs in general have a collective frequency greater than 10^{-4} per year. It is noted CDF is a risk metric that has been defined in PRA standards [31] and regulatory guides [37] in terms that are only meaningful for LWRs. Hence, CDF as a risk metric is not applicable to advanced non-LWRs. However it provides a measure of the design objective for the frequency of beyond design basis accidents[8].

In the NRC Safety Goal Policy, the following performance guideline on the frequency of a large release is provided:

"Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation."

Although this large release frequency (LRF) metric is framed in terms of reactors with LWR containment systems, it is not unreasonable to conclude that the frequency of a large release can be used as a reactor technology metric, so long as the performance objective for a large release is defined. A proposed risk metric for demonstrating compliance with the Commission's LRF goal is included as part of the LMP LBE evaluation criteria as discussed in Section 3.

The NRC has not established a lower bound for the frequency of severe accidents that need to be considered. However, in general, the NRC does not require consideration of accidents that are not deemed to be 'credible.' Additionally, Regulatory Guide 1.174, Section A.2.4 [36], states that an increase in core damage frequency of less than 10^{-6} per year and an increase in large early release frequency of $<10^{-7}$ per year are considered 'very small' and consistent with the Commission's Safety Goal Policy. These criteria are repeated in Section III.2.2.5 of SRP 19, "Use of Probabilistic Risk Assessment in Plant-Specific Risk-Informed Decision-making: General Guidance." Additionally, SRP 19 states that a PRA may have a truncation limit that, depending on the level of PRA detail (module level, component level, or piece-part level), may be from 10^{-12} to 10^{-8} per reactor-year. Similarly, Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities," Section 1.2.5 [37], states that an external event may be screened out of a PRA if it can be

[8] Although some degree of core damage is assumed in the formulation of design basis source terms for evaluating containment leak rate and siting criteria, severe core damage phenomena such as direct containment heating, hydrogen conflagrations, steam explosions, over-pressurization, containment bypass, and other phenomena known to contribute to containment failure probability are explicitly considered as part of the design basis.

shown that the mean value of the frequency of the corresponding design-basis hazard used in the plant design is less than 10^{-5} per year and that the conditional core-damage probability is less than 10^{-1}, given the occurrence of the design-basis hazard. These guidelines indicate that events that have a frequency lower than ~10^{-6} or 10^{-7} per year do not need to be evaluated, and that events with a frequency of less than about 10^{-8} may be screened from the PRA.

A.4.5 United Kingdom Safety Assessment Principles (SAPs) Numerical Targets

As previously discussed the UK SAPs appear to roughly correspond to a collection of U.S. regulatory requirements, General Design Criteria, regulatory guides, standard review plan, and safety goals. The SAPs call for a fault analysis to be performed first for each separate nuclear facility that is submitted for a Generic Design Assessment and then for the site as a whole. The fault analysis supports three types of analyses that are expected to be done in an integrated fashion and in a manner that complements each other. The analyses are referred to as Design Basis Accident Analysis, PSA[9], and Severe Accident Analyses. The analyses are performed against a set of deterministic requirements set forth in the SAPs as well as numerical targets on the frequencies, consequences, and risks to individuals on-site and off-site. The numerical risk targets reflected in the SAPs are listed in **Table** 6.

There is a long history in the development of numerical risk targets for evaluating nuclear reactor safety which started with a paper by Reginald Farmer [64]. Dr. Farmer was actually responsible for performing the first PSA of reactor accidents in the 1960's which predated by a decade the first LWR PRA in WASH-1400. What is significant to appreciate about the UK experience with the development of the SAPs is that they are reactor-technology neutral, having been applied to two different types of graphite moderated gas-cooled reactors, an operating LWR at Sizewell, and current being used to evaluate a number of ALWR designs as well as several liquid metal cooled fast reactor designs.

The numerical risk targets reflected in the SAPs provide extremely useful guidance in the development of the proposed TLRC as explained more fully in Section 3 of this paper.

[9] IAEA and SAP documents use the term Probabilistic Safety Analysis (PSA) to mean the same thing as Probabilistic Risk Analysis (PRA) according to the IAEA safety term glossary in Reference [38].

Table 6 United Kingdom Safety Assessment Principles Numerical Risk Targets [39]

No.	Applicable State or Event	Applicable to	Facility or Site Based	Basic Safety Objective (BSO)	Basic Safety Limit (BSL)	Applicable Event Frequency or Consequence
1	Normal Operation	Any person on site (Radiation worker)	Site	< 1mSv/year	< 20mSv/year	Annual limits
		Any person on site (Other employees)	Site	< 0.1mSv/year	< 2mSv/year	
2	Normal Operation	Any group on site (Radiation workers)	Site	< 0.5mSv/year	< 10mSv/year	
3	Normal Operation	Any person off site	Site	0.02mSv/year	1mSv/year	
4	Design Basis Accidents	Any person on site	Facility	< 0.1mSv/event	20mSv/event	> 10^{-3}/year
					200mSv/event	10^{-3} to 10^{-4}/year
					500mSv/event	10^{-4} to 10^{-5}/year
		Any person off site		< 0.01mSv/event	1mSv/event	> 10^{-3}/year
					10mSv/event	10^{-3} to 10^{-4}/year
					100mSv/event	10^{-4} to 10^{-5}/year
5	All accidents	Any person on site	Site	< 10^{-6} /year	< 10^{-4} /year	fatality
6	All accidents	Any Person on site	Facility	< 10^{-3}/year	< 10^{-1}/year	2-20mSv
				< 10^{-4}/year	< 10^{-2}/year	20-200mSv
				< 10^{-5}/year	< 10^{-3}/year	200-2000mSv
				< 10^{-6}/year	< 10^{-4}/year	> 2,000mSv
7	All accidents	Any person off-site	Site	< 10^{-6} /year	< 10^{-4} /year	fatality
8	All accidents	Any Person off site	Facility	< 10^{-2}/year	< 1/year	0.1-1mSv
				< 10^{-3}/year	< 10^{-1}/year	1-10mSv
				< 10^{-4}/year	< 10^{-2}/year	10-100mSv
				< 10^{-5}/year	< 10^{-3}/year	100-1,000mSv
				< 10^{-6}/year	< 10^{-4}/year	> 1,000mSv
9	All accidents	All persons on and off-site	Site	< 10^{-7}/year	< 10^{-5}/year	≥ 100 early or latent fatalities

A.5 Regulatory Foundation Precedent Review Summary

The following observations and conclusions are made in this review of the regulatory foundation for selection of LBEs for advanced non-LWRs. These observations and conclusions shape the development of an approach for LBE selection that is provided in Section 3 of this white paper.

- Existing NRC Policy and Strategy statements fully support the greater use of RIPB practices. This vision is clearly articulated in NUREG-2150. There has been partial development of RIPB methods for the backfit, operation, oversight and modification of existing LWRs, however, little or no guidance for RIPB decision making has been established for new, non-LWR advanced designs.
- The current U.S. regulations and regulatory guidance ("framework") for LWR-based designs do not include or provide a reproducible approach for selecting LBEs for advanced non-LWRs nor for ensuring that advanced non-LWRs of differing designs would be treated in a consistent manner for establishing their design and licensing bases.
- The only reactor technology inclusive set of regulatory documents that was identified in this regulatory review is that reflected in the U.K. Safety Assessment Principles (SAPs). The SAPs include numerical targets for evaluating LBE frequencies and consequences which differentiate between those to be applied to each reactor unit and those which apply to the site as a whole. Different targets are expressed for regulatory evaluation boundaries and design objectives, thereby capturing the notion that risk are not to be used a strict pass-fail acceptance test.
- , PBMR and NGNP project, as well as the approach used for PRISM for LBE selection, provide an appropriate baseline from which to develop the LBE selection process for advanced reactor design and licensing. An LBE selection approach proposed in NUREG-1860 was also reviewed for insights to help define desirable attributes of an effective LBE selection process. This regulatory foundation review provides guidance for refining and advancing these approaches.
- The RIPB approach advanced in the MHTGR, PBMR and NGNP projects has been reflected in a design standard for MHRs in ANS 53.1. This standard provides specific design criteria for implementing the approach that is consistent with the approach described in the NGNP white papers. These include criteria for evaluating the adequacy of defense-in-depth which contributes to the deterministic input to RIPB design decisions.
- There are a number of international precedents, including those from the U.S., IAEA, and the U.K. SAPs, and reflected in the NRC reviews of MHTGR, PRISM, and NGNP, that support the view that LBE selection is best accomplished through a risk-informed and performance-based process which includes both deterministic and probabilistic inputs and preserves the principle of defense-in-depth.
- A key challenge of any LBE selection process is to systematically define the initiating events that are appropriate for the reactor design, and the event sequences that realistically model the plant response to the initiating events. This is necessary in order to derive the appropriate and limiting Design Basis Accidents (DBAs) for that design. Simply removing inapplicable events from existing LWR events is not sufficient to define the events that are uniquely appropriate for a given design.
- The LBE definition and selection process must be clear in making the distinction between initiating events and event sequences. A given initiating event may result in different event sequences each having a different frequency of occurrence and level of severity in challenging the reactor safety defenses. Simply assuming the "worst active single failure" and concurrent loss of offsite power in combination with an initiating event does not necessarily yield the appropriate limiting accidents to define the licensing basis.
- As emphasized in NUREG-1860, PRA plays an important role in the identification and evaluation of uncertainties in the definition of event sequences and in the estimation of their frequencies and consequences. This information on sources of uncertainty and their influences on the risk

assessment are important inputs to establishing adequate consideration of the principles of defense-in-depth in the selection and evaluation of LBEs and other RIPB decisions.

- In order to provide the technical basis for managing the risks of accidents that involve two or more reactors or radionuclide sources, by preventing and mitigating such accidents, it is necessary to consider such accidents in the definition of LBEs and to measure frequencies on a per (multi-reactor module) plant[10]-year basis, rather than reactor-year basis.
- The development of TLRC frequency-consequence criteria for the LMP project greatly benefits from the approach most recently advanced in the NGNP LBE white paper as well as similar frequency-consequence criteria originally proposed by Farmer. Useful guidance is also available from NUREG-1860 the U.K. SAPs for event consequences, frequencies and threshold for event evaluation.
- A key challenge in interpreting the current U.S. regulations for limiting radiological exposures for normal operation and LBEs is the lack of explicit numerical criteria for categorizing events by expected frequency of occurrence. However, the classification of LBEs into Anticipated Operational Occurrences (AOOs), Design Basis Events (DBEs), and Beyond Design Basis Events (BDBEs) based on expected frequency of occurrence is consistent with LBE classifications that were identified in this regulatory review including NGNP, PRISM, NUREG-1860, NUREG-2150, Yucca Mountain Pre-closure Safety Analysis, and the U.K. SAPs.
- There are a number of NRC criteria that explicitly constrain the risk and/or allowable consequences of radiological releases from nuclear power plants. These criteria include requirements to evaluate the adequacy of the proposed design of the plant against specific limits. Some of the regulatory dose requirements are intended for evaluation of individual events, whereas others are expressed in terms of annual exposure limits, frequency of a given magnitude of release, and individual risks for the population in the vicinity of the plant site. The review of these criteria that was performed in the NGNP LBE White Paper [1] has been extended in this white paper and has yielded some new insights that are reflected in the proposed LBE selection approach as discussed in the next section.

The above key points have been used to guide the development of the LBE selection process as discussed more fully in Section 3.

[10] *Plant*, as the term is used in this document means a nuclear plant that may or may not employ a *modular design*.

Modular design means a nuclear power station that consists of two or more essentially identical nuclear reactors (modules) and each module is a separate nuclear reactor capable of being operated independent of the state of completion or operating condition of any other module co-located on the same site, even though the nuclear power station may have some shared or common systems [2].

A.6 References

[1] Idaho National Laboratory, “Next Generation Nuclear Plant Licensing Basis Event Selection White Paper”, INL/EXT-10-19521, September 2010, [ADAMS Accession No. ML102630246].

[2] 10 CFR Part 52.1, “Licenses, Certifications, and Approvals for Nuclear Power Plants – Definitions”, 2015

[3] NUREG-0800, Standard Review Plan, Chapter 15.0, “Introduction - Transient and Accident Analyses”, Revision 3 March 2007

[4] 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Code of Federal Regulations, Office of the Federal Register, 2007.

[5] 10 CFR 50.62, “Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants”, 2015

[6] 10 CFR 50.63, “Loss of all alternating current power”, 2015

[7] 10 CFR 50.44, “Combustible gas control for nuclear power reactors”, 2015

[8] 10 CFR 50.150, “Aircraft impact assessment”, 2015

[9] U.S. Nuclear Regulatory Commission, ‘Policy Statement on the Regulation of Advanced Reactors”; 73 FR 60612, October 14, 2008

[10] U.S. Nuclear Regulatory Commission, “Policy Statement on Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities” 60 FR 42622, August 16, 1995

[11] U.S. Nuclear Regulatory Commission, ‘Policy Statement on Safety Goals for Operation of Nuclear Power Plants”, 51 FR 28044 August 21, 1986

[12] U.S. Nuclear Regulatory Commission, “Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants”; 50 FR 32138, August 8, 1985

[13] SECY 2002-0076, “Semi-Annual Update of the Future Licensing and Inspection Readiness Assessment (FLIRA)”, May 8, 2002

[14] SECY 2003-0047, "Policy Issues Related to Licensing Non-Light Water Reactor Designs," U.S. Nuclear Regulatory Commission, March 28, 2003.SECY 2002-0139

[15] SRM 2003-0047, "Staff Requirements Memorandum for SECY 03-0047-Policy Issues Related to Licensing Non-Light Water Reactor Designs," U.S. Nuclear Regulatory Commission, June 26, 2003.

[16] SECY-2005-0006, “Second Status Paper on the Staff's Proposed Regulatory Structure for New Plant Licensing and Update on Policy Issues Related to New Plant Licensing”, January 7, 2005

[17] SECY-2010-0034, “Potential Policy, Licensing, and Key Technical Issues for Small Modular Nuclear Reactor Designs”, March 28, 2010

[18] SECY-2011-0079, “License Structure for Multi-Module Facilities Related to Small Modular Nuclear Power Reactors”, June 12, 2011

[19] NUREG-0800, Standard Review Plan, Chapter 19.0, “Probabilistic Risk Assessment and Severe Accident Evaluation For New Reactors”, Revision 3 December 2015

[20] SECY 2016-0012, “Accident Source Terms and Siting for Small Modular Reactors and Non-Light Water Reactors”, February 7, 2016

[21] NUREG-1860, “Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing”, U.S. Nuclear Regulatory Commission, December 2007.

[22] NUREG-2150, “A Proposed Risk Management Regulatory Framework”, U.S. Nuclear Regulatory Commission, April 2012

[23] U.S. Nuclear Regulatory Commission, “The Near-Term Task Force Review of Insights From the Fukushima Dai-Ichi Accident”, July 12, 2011

[24] NUREG/BR-0303, Guidance for Performance Based Regulation, U.S. Nuclear Regulatory Commission, December 2002

[25] ANS-53.1-2001, “Nuclear Safety Design Process for Modular Helium-Cooled Reactor Plants”, American Nuclear Society, December 2011

[26] Idaho National Laboratory, “Next Generation Nuclear Plant Defense-in-Depth Approach White Paper”, INL/EXT-09-17139, March 2009, [ADAMS Accession No. ML102630246].

[27] Department of Energy, “Yucca Mountain Repository Safety Analysis Report” Chapters 1.6 - 1.9 and 1.14, Docket No. 63–001, DOE/RW-0573, Rev. 1, February 2009

[28] NUREG-2108, “Technical Evaluation Report on the Content of the U.S. Department of Energy’s Yucca Mountain Repository License Application: Preclosure Volume: Repository Safety Before Permanent Closure”, Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, September 2011

[29] 10 CFR Part 63, “Disposal Of High-Level Radioactive Wastes In a Geologic Repository at Yucca Mountain, Nevada”, 2016

[30] NUREG-1804, “Yucca Mountain Review Plan”, Revision 2, July 2003

[31] ASME/ANS RA-Sb-2013, “Addenda to ASME/ANS RA-S–2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications”, American Society of Mechanical Engineers and American Nuclear Society, September 2013

[32] ASME/ANS RA S-1.4-2013, “Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants”, American Society of Mechanical Engineers and American Nuclear Society, December 2013

[33] 10 CFR Part 63.111, “Performance objectives for the geologic repository operations area through permanent closure”, 2015

[34] 10 CFR Part 63.112, “Requirements for preclosure safety analysis of the geologic repository operations area”, 2015

[35] SRM 90-16, “Staff Requirements Memorandum for SECY 90-16 Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements”, January 1990

[36] Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment i_n Risk-informed Decisions on Plant Specific Changes to the Current Licensing Basis," Federal Register, U.S. Nuclear Regulatory Commission, Revision 2, May 2011

[37] Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities," Federal Register, U.S. Nuclear Regulatory Commission , Revision 2, March 2009.

[38] International Atomic Energy Agency, “Safety Standards Series: Safety of Nuclear Power Plants Design Requirements”, NSR-1, September 2000

[39] United Kingdom Office of Nuclear Regulation, “Safety Assessment Principles for Nuclear Facilities”, 2014 Edition Revision 0

[40] U.S. Department of Energy, Preliminary Safety Information Document for the Standard MHTGR, DOE-HTGR-86-024, September 1988.

[41] U.S. Department of Energy, Probabilistic Risk Assessment for the Standard Modular High Temperature Gas-Cooled Reactor, DOE-HTGR-86-011, Revision 5, April 1988.

[42] NUREG-1338, "Draft Preapplication Safety Evaluation Report for the Modular High Temperature Gas-Cooled Reactor," U.S. Nuclear Regulatory Commission, March 1989.
[43] NUREG-1368, Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor, Final Report, U.S. Nuclear Regulatory Commission, February 1994
[44] Edward G. Wallace letter to NRC Document Control Desk, "PBMR White Paper: LBE Selection," July 3, 2006.
[45] NRC letter to PBMR, "Requests For Additional Information Regarding Pebble Bed Modular Reactor (PBMR) Pre-Application White Papers," September 24, 2007.
[46] PBMR Limited letter to NRC, "Response to Requests for Additional Information," March 21, 2008.
[47] Exelon Generation Company Letter, Subject: Proposed Licensing Approach for the Pebble Bed Modular Reactor in the United States, January 31, 2002.
[48] U.S. Nuclear Regulatory Commission Letter, Subject: "NRC Staff's Preliminary Findings Regarding Exelon Generation's (Exelon's) Proposed Licensing Approach For The Pebble Bed Modular Reactor (PBMR)," March 26, 2002.See also SECY-95-0299, "Licensing Approach for Exelon's Pebble Bed Modular Reactor", December 19, 1995.
[49] INL/EXT-13-28205," NRC Licensing Status Summary Report for NGNP", Idaho National Laboratory, November 2014.
[50] Letter from J. Sam Armijo, Chairman U.S. Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards to Mr. R.W. Borchardt, Executive Director for Operations, U.S. Nuclear Regulatory Commission Letter, "Next Generation Nuclear Plant (NGNP) Key Licensing Issues", May 15, 2013
[51] SECY 2011-0152, "Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors", October 28, 2011
[52] 10 CFR Part 20, "Standards for Protection against Radiation Subpart C, Occupational Dose Limits", 2016
[53] 10 CFR Part 20, "Standards for Protection against Radiation Subpart D, Radiation Dose Limits for Individual Members of the Public", 2016
[54] 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents", 2015
[55] 10 CFR Part 52, Licenses, Certifications, And Approvals For Nuclear Power Plants, Subpart C "Combined Licenses", 2016
[56] 10 CFR Part 52.79, "Contents of applications; technical information in final safety analysis report", 2015
[57] 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations", 2016
[58] 10 CFR Part 100, "Reactor Site Criteria (Subpart B, Evaluation Factors for Stationary Power Reactor Site Applications on or After January 10, 1997)":
[59] 10 CFR §50.34, "Contents of Applications: Technical Information - Radiological Dose Consequences", 2015
[60] 10 CFR Part 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors", 2015
[61] 40 CFR Part 190.10, "Standards for normal operations", 2016

[62] Idaho National Laboratory, "Next Generation Nuclear Plant Probabilistic Risk Assessment White Paper", INL/EXT-11-21270, Idaho National Laboratory, September 2011.
[63] NUREG-0800, Standard Review Plan, Chapter 15.0.3 "Design Basis Accident Radiological Consequence Analyses For Advanced Light Water Reactors", March 2007
[64] Farmer, F., "Reactor Safety and Siting: A Proposed Risk Criterion", *Nuclear Safety* 1967; pp. 539–48.
[65] NUREG-0800, Standard Review Plan, Chapter 15.0 "Introduction - Transient and Accident Analyses", Revision 3 March 2007
[66] American Nuclear Society, ANSI/ANS-53.1-2011, "Nuclear Safety Design Process for Modular Helium-Cooled Reactor Plants", December 21, 2011
[67] NEDO-33201, Revision 6, ESBWR Design Certification Probabilistic Risk Assessment, Chapter 10.4 COMPARISON OF RESULTS TO GOALS, (Adams Accession No. ML102880536)
[68] U.S. Department of Energy, Licensing Basis Events for the Standard MHTGR, DOE-HTGR-86-034, February 1987.
[69] GE Hitachi Nuclear Energy, "Final Scientific/Technical Report: Development/Modernization of an Advanced Non-LWR Probabilistic Risk Assessment," Federal Grant DE-NE0008325, 2017.
[70] Letter from Mario V. Bonaca, Chairman U.S. Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards to Honorable Nils Diaz, Chairman, U.S. Nuclear Regulatory Commission Letter, "Options and Recommendations for Policy Issues Related to Licensing Non-Light Water Reactor Designs", April 22, 2004
[71] SECY 2013-0029, "History of The Use and Consideration of the Large Release Frequency Metric By The U.S. Nuclear Regulatory Commission", March 22, 2013