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

Selection of Licensing Basis Events
Draft Report Revision 0

Prepared for the
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Office of Nuclear Energy
Under DOE Idaho Operations Office
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Note that the NRC staff comments within this draft document are intended to be considered along with the high-level comments provided. There may be other changes needed within the draft to ensure consistency and to reconcile the draft with the resolution of comments or response to questions.

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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)** **[this terminology is different than current Part 50 definitions and so may be appropriate to add footnote or otherwise make clear that this approach is using same term but with different meaning]** 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 5×10^{-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.

The events evaluated within these categories are used to support various regulatory decisions associated with the design, operation, and siting non-LWR plants.

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** ["risk significant" is

[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.

defined in ASME/ANS RA Sa-2009 and so may need to confirm that terminology is consistent or revise the terminology in this document to avoid confusion] 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.

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** [may wish to refer to ongoing ARDC activities] 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. **The licensing basis events and associated design features included to address them also support performing environmental reviews and establishing necessary operational limits, siting constraints, and appropriate emergency planning requirements.**

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. [Somewhere should address that research, modeling, and other activities provide sufficient understanding of plant and fuel performance to enable performing PRA and deterministic analyses]

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.

Consider additional box to consider outcomes in terms of emergency planning and siting

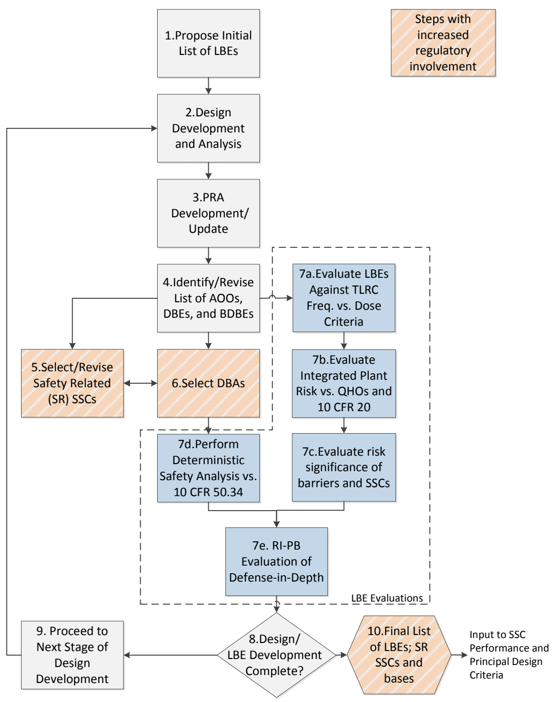

Figure ES-1 Process for Selecting and Evaluating Licensing Basis Events

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 using conservative or best estimate approaches with appropriate accounting for uncertainties.

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~~Correlation [staff comments within paper revised criteria to correlation but other terminology may be preferable (guidelines, reference values, etc.)]

- In this task the results of the PRA which have been organized into LBEs will be ~~evaluated against~~assessed using the ~~TLRC~~ frequency-consequence ~~criteria~~ correlation shown in ~~of~~ Figure ES-2. **The figure does not define specific acceptance criteria for the analysis of LBEs but provides a tool to focus the attention of the designer and those reviewing the design and related operational programs to the most significant events and possible means to address those events. The NRC's Advanced Reactor Policy Statement includes expectations 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. The safety margin between the design-specific PRA results, and the Frequency-Consequence guideline can provide one useful and practical demonstration of how the design fulfills the Commission expectations for enhanced safety.** 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~~limiting the frequency-dose ~~criteria~~ results, 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: [staff suggests deleting criteria related to Part 20 and 750 rem]

- ~~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. Deterministic analyses often use stylized scenarios for the purpose of demonstrating compliance with specific requirements, establish safety margins, and define equipment specifications and operational limits. Deterministic safety analysis can use conservative or best-estimate analytical methods with an appropriate accounting of uncertainties. 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.

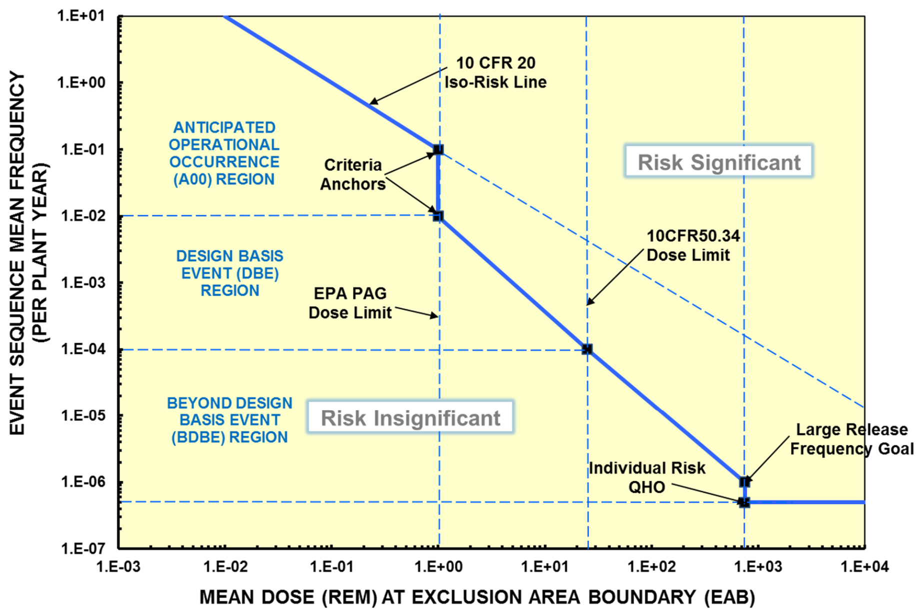

Figure ES-2 Frequency-Consequence Evaluation Criteria Proposed for LMP

Suggest "Decreasing Risk" and "Increasing Risk", with Arrows versus terms implying acceptability or significance. In line with treatment as a tool and not using lines as 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~~limit LBE frequencies and doses~~ within the TLRC frequency-dose criteria~~. Important information from Task 7b is used for this purpose. [Suggest adding that the results of the event selection process and related evaluations will be described in appropriate sections of the SAR.]

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

[The staff did not review in detail those parts of the paper dealing largely with background or historical information. While relevant to the white paper (in terms of context), it is not expected that much of this information would be incorporated into the subsequent consolidated guidance document. Please point out if there is a key point for the subsequent guidance document within any of the text marked in green italics.]

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

[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).

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~~Figure 1-1Figure 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, 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.

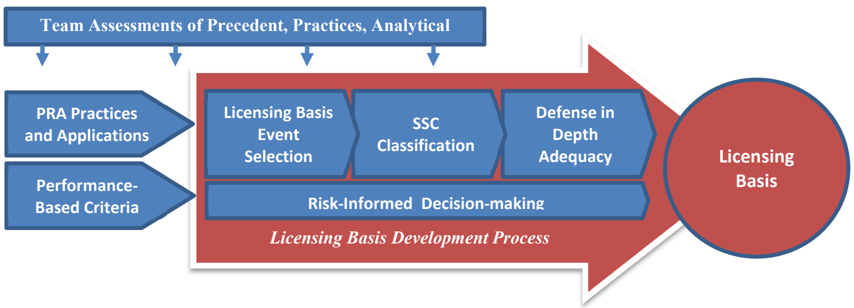

Figure 1-1 Elements of TI-RIPB Licensing Modernization Framework

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). The licensing basis events and associated design features included to address them also support establishing necessary operational limits, siting constraints, and appropriate emergency planning requirements. 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.

Note that specific limits in objectives should reflect any revisions made in response to interactions

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:

- 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~~ three categories of LBEs that can be addressed through the PRA or alternative risk assessment methodologies that involve conservative or best estimate approaches with appropriate accounting for uncertainties:
    - 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, to support development of various mitigating strategies, and to provide input to the selection of DBAs.

The analyses of AOOs, DBEs, and BDBEs using a plant PRA or risk assessment methodologies can also used to assess potential accident consequences to support environmental reviews and decisions related to siting (e.g., 10 CFR 100.21, "Non-seismic site criteria**").**

The fourth category of LBE, assessed using deterministic approaches, are:

- 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). Alternatively, a

[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].

designer may provide confidence DBAs do not result in offsite doses by showing that fission product barriers maintain integrity and thereby prevent the release of radioactive material. 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. An example of using best estimate approaches with an accounting for uncertainties in provided in Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance."

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 or other risk assessment methodology 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], [Note that non-reactor sources may be significant contributors for some advanced reactor technologies and so this point could be emphasized more than mention in a footnote] 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.

[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.

- 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~~ Event sequence consequences for the LBE categories ~~are defined by~~ can be assessed by showing fission product barriers maintain integrity and thereby prevent the release of radioactive material or using a frequency-consequence evaluation~~-~~to identify design and related operational programs needed to provide reasonable assurance of adequate protection of public health and safety or a cost-effective approach to address BDBEs. ~~criteria derived from Top Level Regulatory Criteria (TLRC)~~). The ~~TLRC~~ frequency-consequence ~~criteria are~~correlation is used to evaluate the risk significance of each LBE. Key ~~elements of the TLRC~~ factors used to develop the frequency-consequence ~~criteria~~ correlation 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 limit for DBAs: 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 (note that distances are likely to be OK, will need to be justified for non-LWR technologies based on release mechanisms and comparison to LLWRs).

  Designers may address design goals related to siting or emergency planning by showing that AOOs, DBEs, and BDBEs would not exceed specific dose acceptance criteria (e.g., 1 rem at the site boundary to limit the requirements related to offsite emergency planning).

- 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.~~
  - 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. Designers may address design goals related to siting or emergency planning within the assumptions included in their risk assessments.

- 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.
- External events may require special consideration for establishing design specifications for structures, systems, and components and their treatment within safety classification and reliability programs. This will be addressed in a subsequent white paper ???

- 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~~ frequencies and consequences of event sequence families are ~~explicitly compared to the consequence criteria in all applicable LBE regions~~ assessed to identify design and related operational programs needed to provide reasonable assurance of adequate protection of public health and safety or a cost-effective approach to address BDBEs.

  - 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
- Mechanistic Source Term ?

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~~Table 2-1~~~~*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

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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~~Table 2-2Table 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 historical information is probably not needed for incorporation into the more consolidated guidance. Given the staff is suggesting that the F-C curve not be presented as criteria, but instead as a tool to focus attention, the derivation of the LMP correlation (e.g., Section 3.2.6) might be provided as background – perhaps along with other representations that define roughly comparable relationships. When not being used as criteria but only as a tool, the roughly comparable figures provide some assurances versus debate about specific points on the figure.

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~~Figure 3-1~~~~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~~Figure 3-2Figure 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.

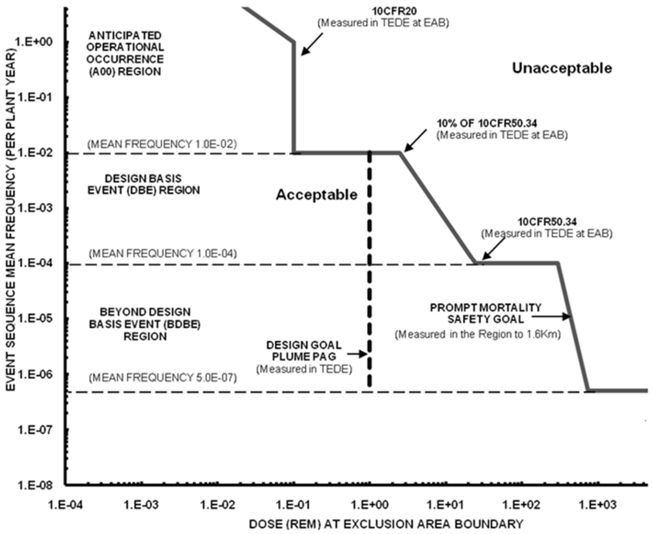

Figure 3-1 NGNP TLRC Frequency – Consequence Criteria

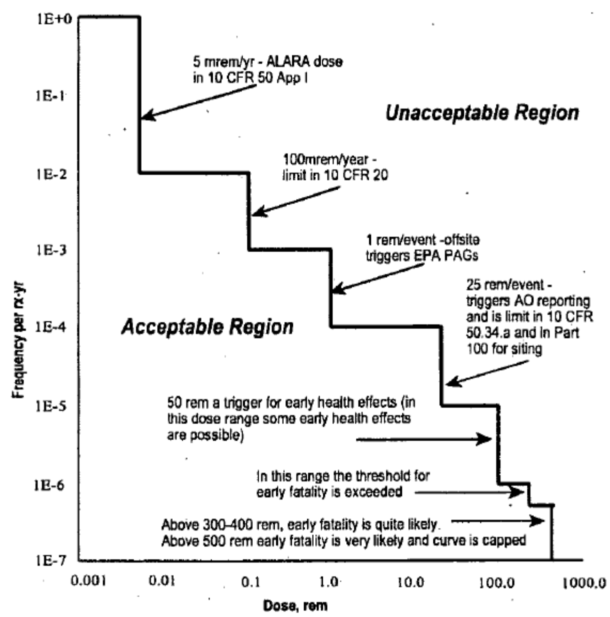

Figure 3-2 NUREG-1860 Frequency – Consequence Criteria

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~~Figure 3-3Figure 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 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.

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

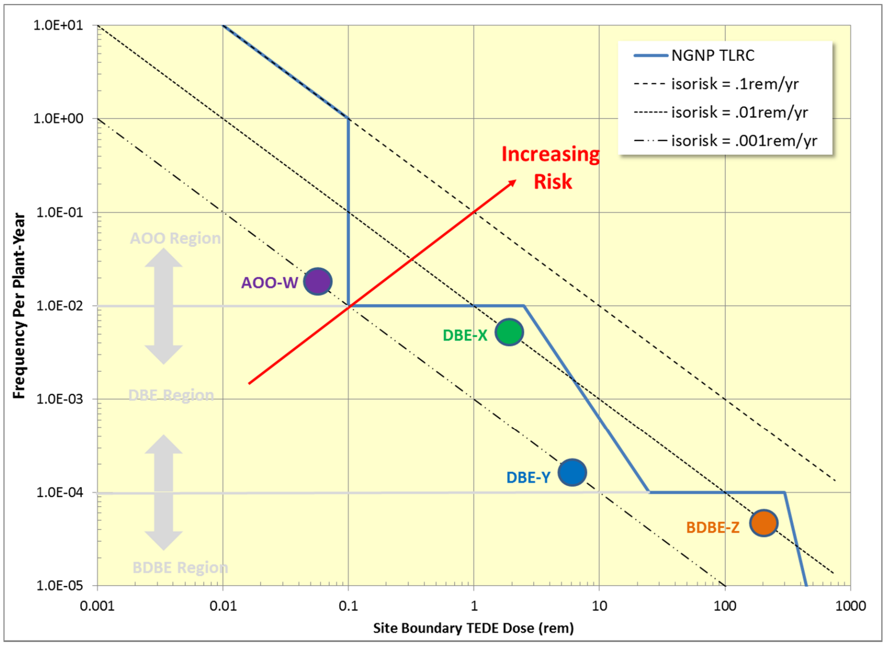

Figure 3-3 Part of NGNP Frequency-Consequence Criteria Illustrating Staircase Issue

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~~Figure 3-4Figure 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.

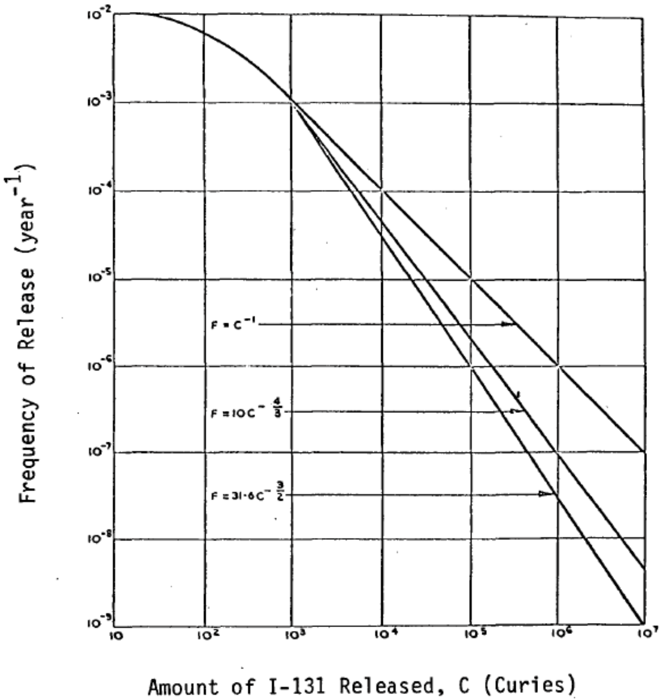

Figure 3-4 Frequency vs. Consequence Limit Line Proposed by Farmer [64]

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~~accounts for 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. Although shown on the F-C graph, developer and users need to keep in mind that the factors are related to the aggregate risks from all LBEs and may not be appropriate for assessing individual LBEs. The integrated risks could, in theory, result in a need to consider further plant improvements or result in limitations related to siting, emergency planning, or operational programs beyond those identified by assessing individual LBEs. 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
- 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~~correlation, and are applied on a per plant-year basis so that the integrated risks of accidents involving single and multiple reactor modules and

~~[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.~~

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~~ correlation 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~~ correlation proposed for the LMP project are shown in Figure 3-5. ~~These~~ This correlation is ~~criteria are~~ based on the following considerations:

- The regions of the graph separated by the frequency-dose evaluation line are identified as "~~risk significant~~"Increasing Risk" rather than "unacceptable", and "~~risk insignificant~~Decreasing Risk" 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. However, other factors may prove to be more restrictive than the F-C correlation because plant responses to AOOs are expected to limit challenges to fission product barriers and support resumption of operation following possible plant repairs or other corrective actions.
- The ~~dose criteria for~~ F-C line 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~~F-C line 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. [Clarify which are event goals and which are integrated goals]
- 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 3-4*~~~~Figure 3-4~~.

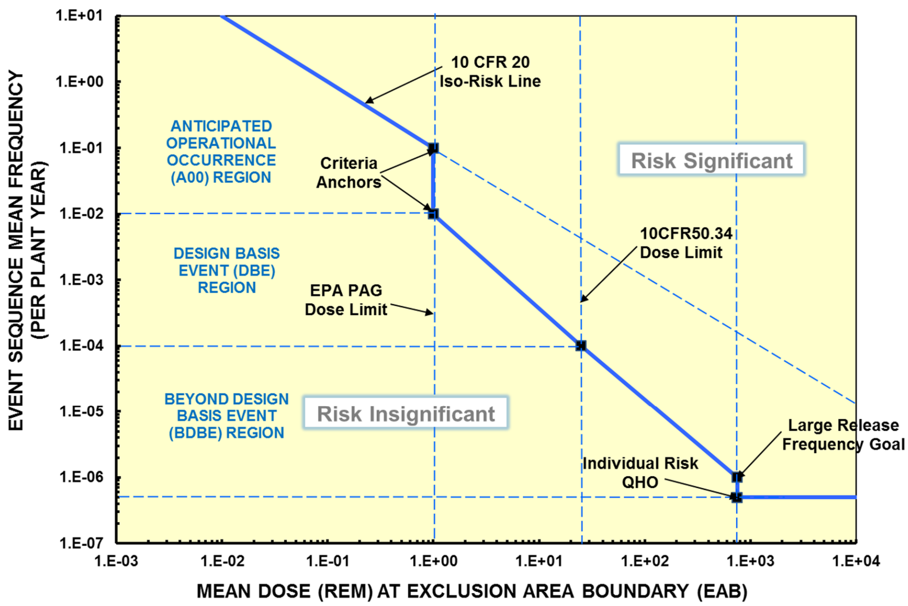

Figure 3-5 Frequency-Consequence Evaluation Criteria Proposed for LMP

The ~~criteria~~ correlation 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~~Figure 3-6Figure 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.

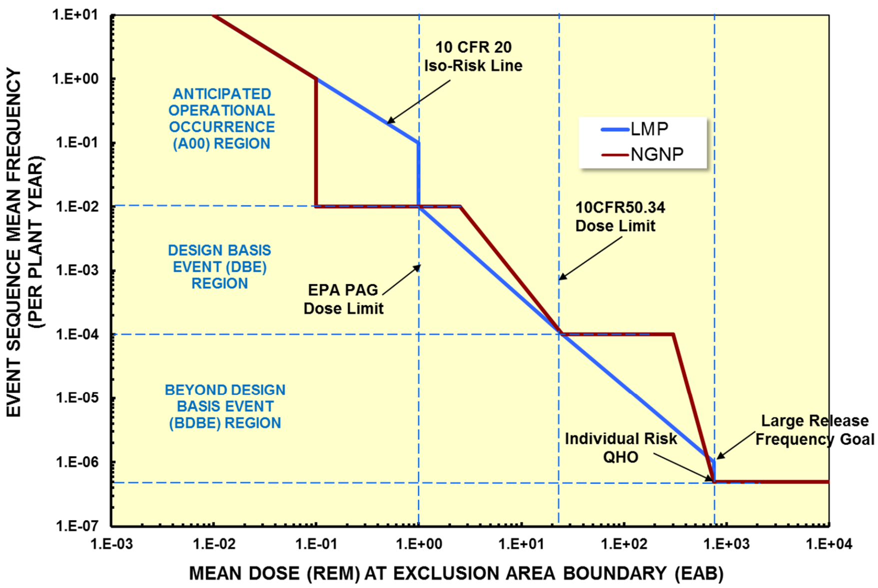

Figure 3-6 Comparison of LMP and NGNP Frequency – Consequence Criteria

A similar comparison is made in Figure 3-7~~Figure 3-7~~~~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.

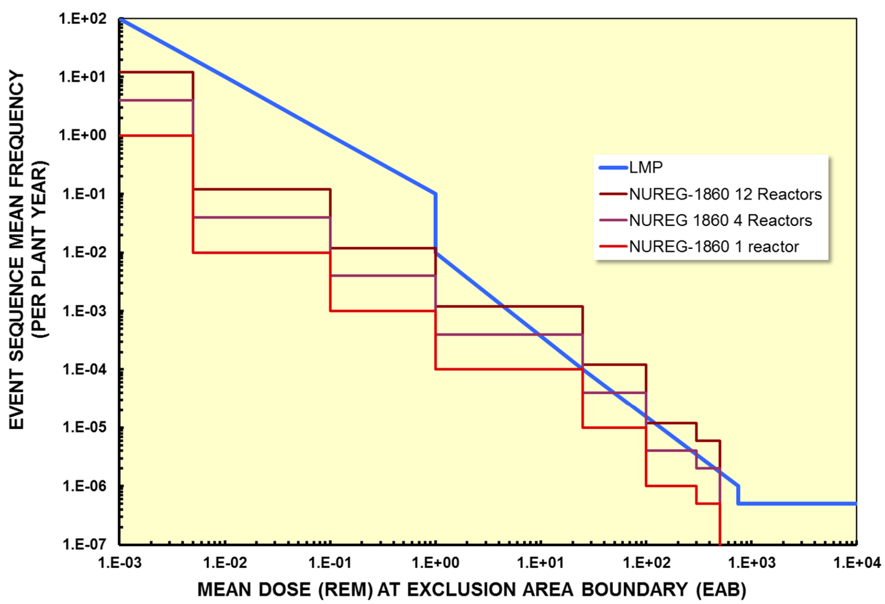

Figure 3-7 Comparison of LMP and NUREG-1860 Frequency – Consequence Criteria

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.

Consider additional box to consider outcomes in terms of emergency planning and siting

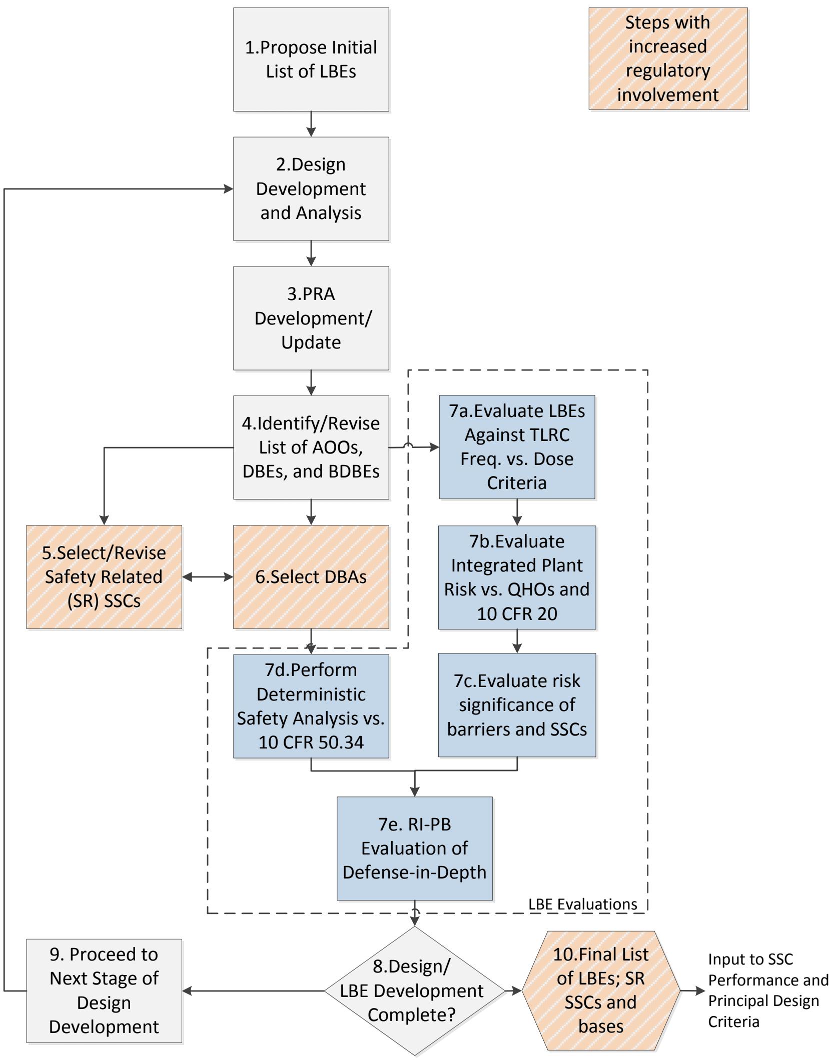

Figure 3-8 Process For Selecting and Evaluating Licensing Basis Events

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 5×10^{-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 5×10^{-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. NRC Regulatory Guide 1.203, "Transient and Accident Analysis Methods," provides additional discussion of developing appropriate evaluation models for analyzing DBAs. 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. [Single unit DBAs seem to imply a certain treatment of external events. Suggest a specific discussion within the guidance related to external events, how approach is similar to or different from traditional "design-basis external events," if approach might include mix of traditional and risk-informed approaches, and implications for safety classification of protections against "design-basis external events."]

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~~Correlation

In this task the results of the PRA which have been organized into LBEs will be evaluated against the ~~TLRC~~ frequency-consequence ~~evaluation criteria~~relationships shown in ~~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.~~ are evaluated and included in discussions of design features and related operational programs related to the most significant events and possible means to address those events. 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~~ correlation 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~~ correlation have been conservatively defined for use as a tool for focusing attention on matters important to managing the risks from non-LWRs. .

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. This may also be a convenient point for designers to assess plant features for effective compliance with regulatory requirements such as 10 CFR 50.155, "Mitigation of Beyond-Design Basis Events," and 10 CFR Part 73, "Physical Protection of Plants and Materials." The results from the evaluation will also support related licensing matters such as defining appropriate constraints in terms of siting (10 CFR Part 100), offsite emergency planning, and development of plant procedures and guidelines.

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, limit the need for restrictions on siting or emergency planning, 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~~ control the LBE frequencies and doses ~~within the TLRC frequency-dose criteria~~or other performance standards associated with the protection of fission product barriers. 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~~ correlation 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.

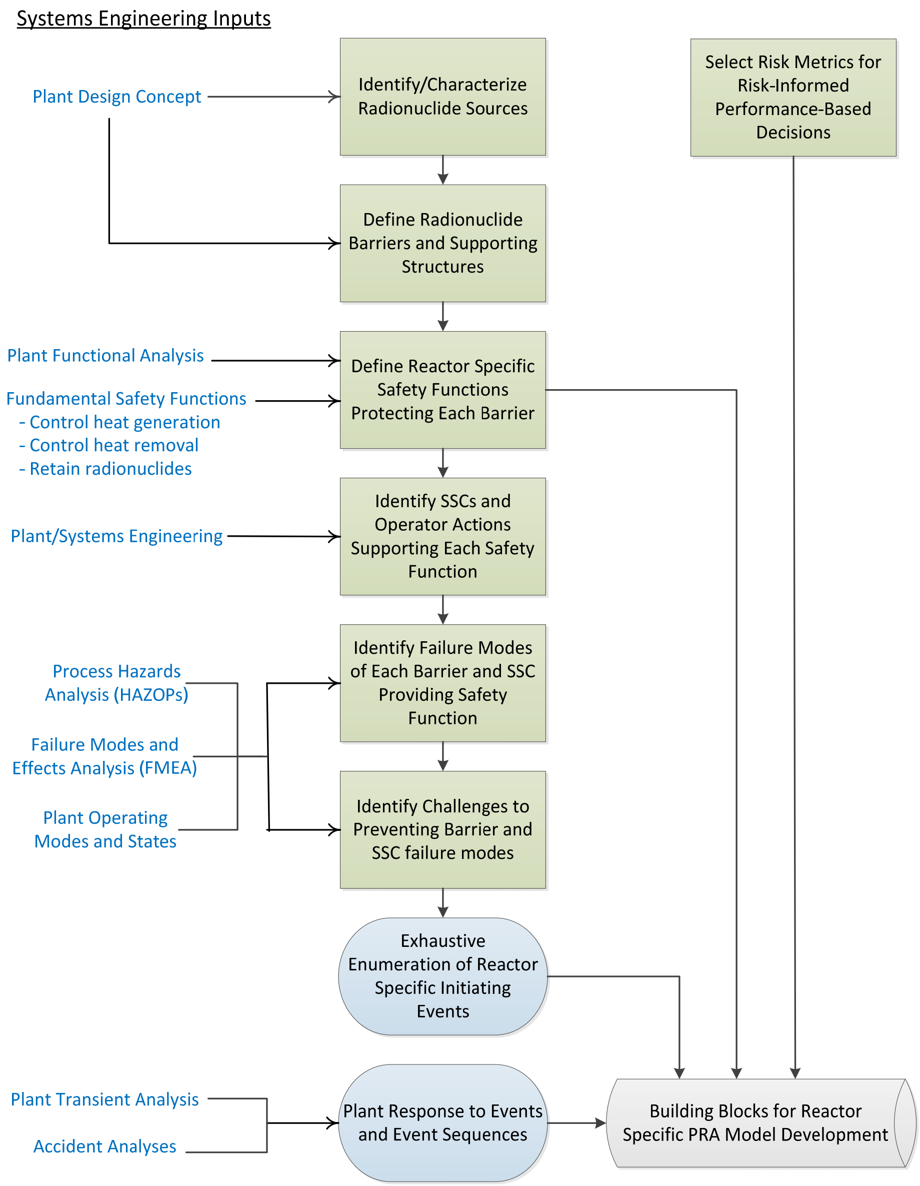

Figure 3-9 Flow Chart for Initial PRA Model Development

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~~F-C correlation. 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. Designers may propose to address all or parts of the process by assessing fission product barriers and showing that radioactive materials are with a high degree of confidence retained within the facility.

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 may include~~s~~ 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.

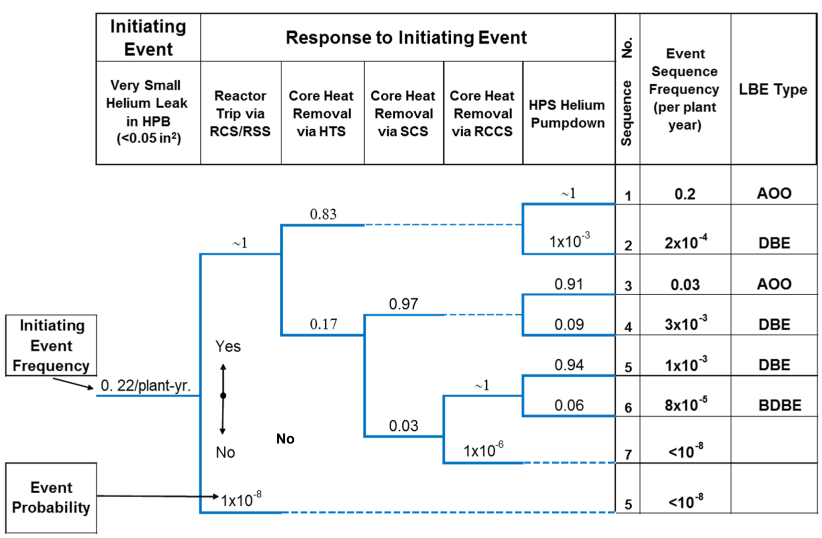

Figure 3-10 Event Tree for MHTGR Very Small Leaks in Helium Pressure Boundary

[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.

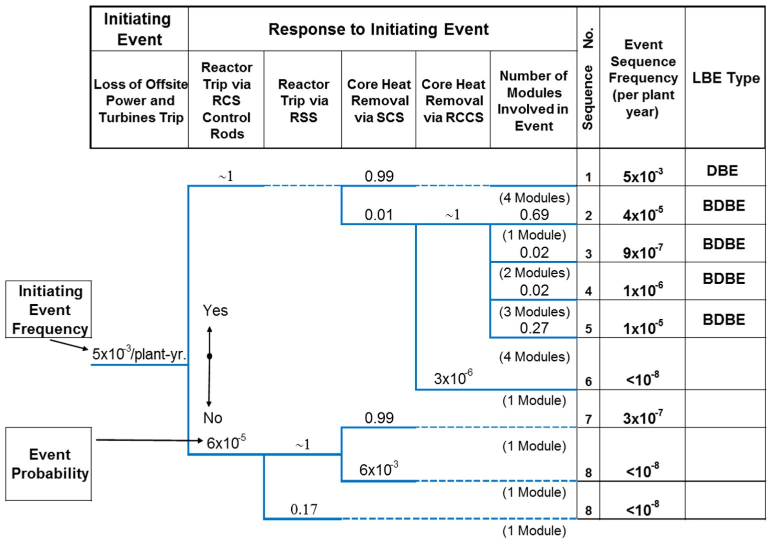

Figure 3-11 Event Tree for MHTGR Loss of Offsite Power and Turbine Trip

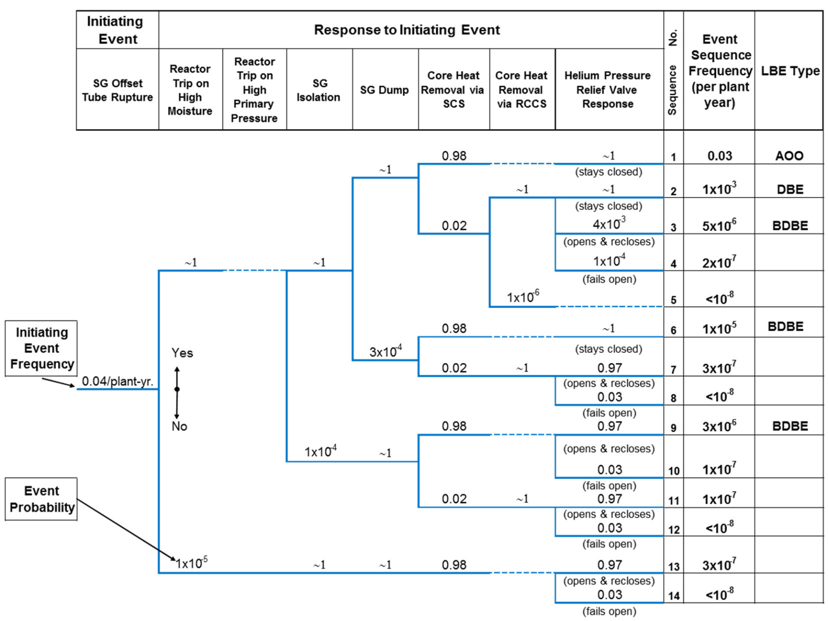

Figure 3-12 Event Tree for MHTGR Steam Generator Tube Rupture

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.

LBE Designation	LBE Description
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.
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.

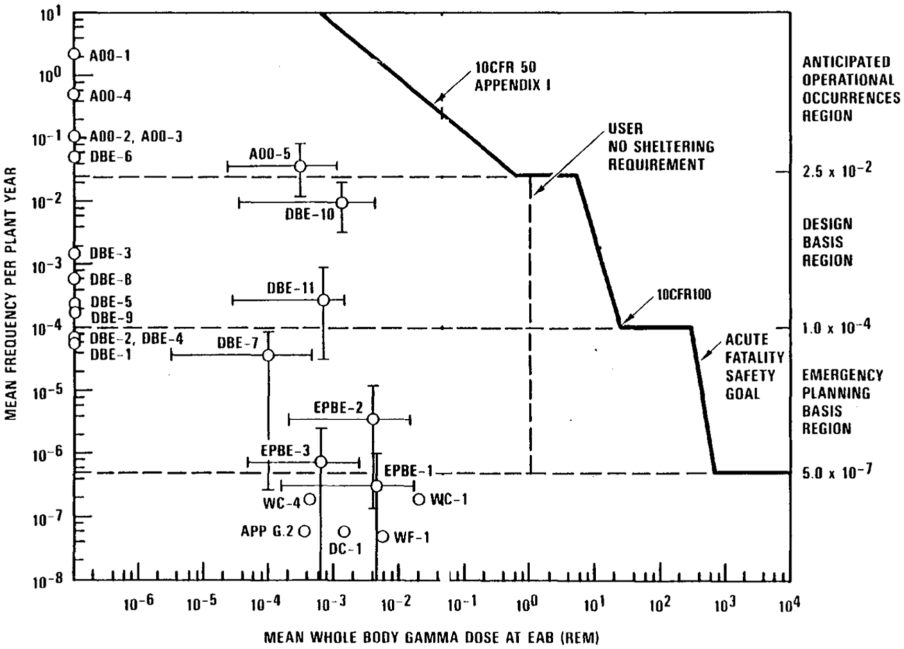

Figure 3-13 Comparison of MHTGR LBE Frequencies and Consequences TLRC Frequency – Dose Criteria[14]

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

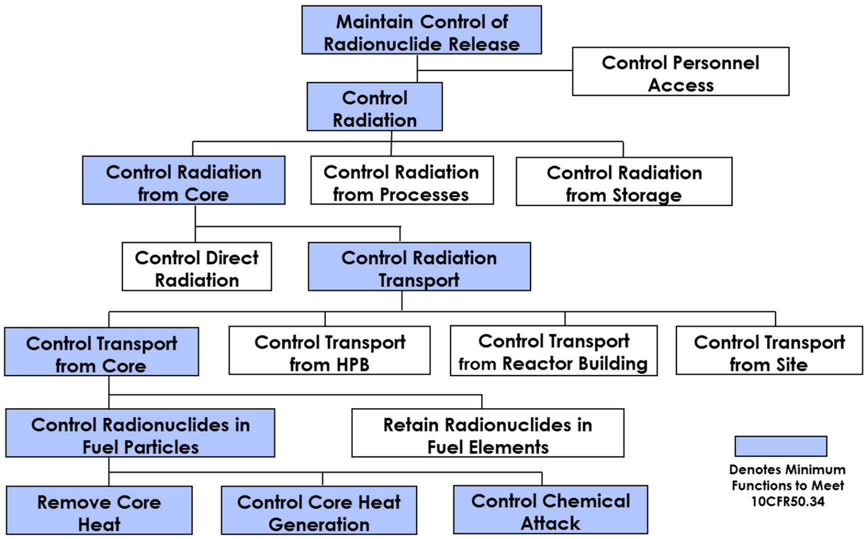

Figure 3-14 MHTGR Safety Functions[15] Including Those Required to Meet 10 CFR 50.34 Limits

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

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[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

DBE	Design Basis Events	DBA	Design Basis Accidents
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. (corresponds to PRA sequence family with frequency of 5x10^{-5}/plant-year or about 1x10^{-5}/reactor-year)	DBA-1	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)
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. (corresponds to PRA sequence family with frequency of 7x10^{-5}/plant-year or about 2x10^{-5}/reactor-year)	DBA-2	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)
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. (corresponds to PRA sequence family with frequency of 2x10^{-3}/plant-year or about 5x10^{-4}/reactor-year)	DBA-3 DBA-4	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)
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. (corresponds to PRA sequence family with frequency of 7x10^{-5}/plant-year or about 2x10^{-5}/reactor-year)		

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)

<table>
<tr><th>DBE</th><th>Design Basis Events</th><th>DBA</th><th>Design Basis Accidents</th></tr>
<tr><td>DBE-7</td><td>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)</td><td rowspan="3">DBA-7
DBA-8
DBA-9</td><td rowspan="3">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)</td></tr>
<tr><td>DBE-8</td><td>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)</td></tr>
<tr><td>DBE-9</td><td>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)</td></tr>
<tr><td>DBE-10</td><td>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)</td><td>DBA-10</td><td>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)</td></tr>
<tr><td>DBE-11</td><td>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</td><td>DBA-11</td><td>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.</td></tr>
</table>

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

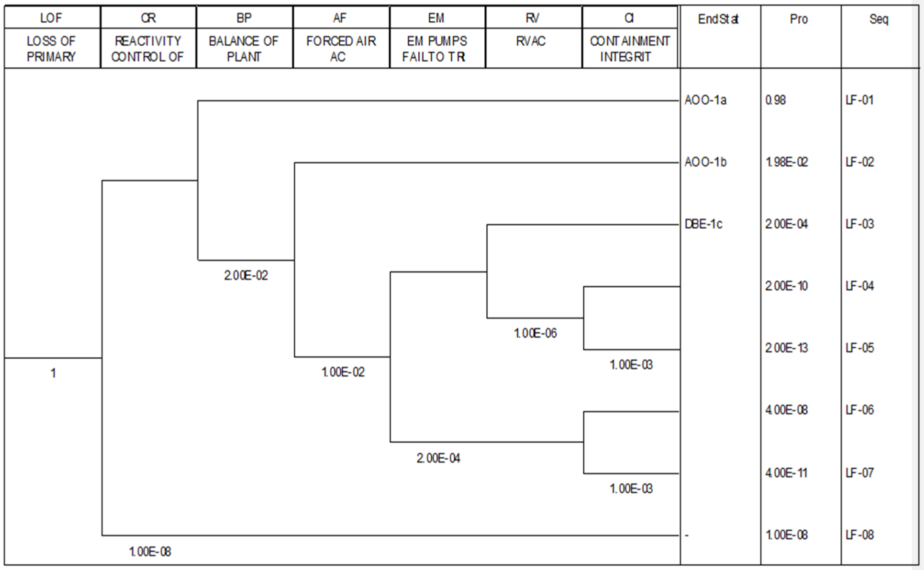

Figure 3-15 Event Tree for Loss of Flow in a Single EM Pump

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

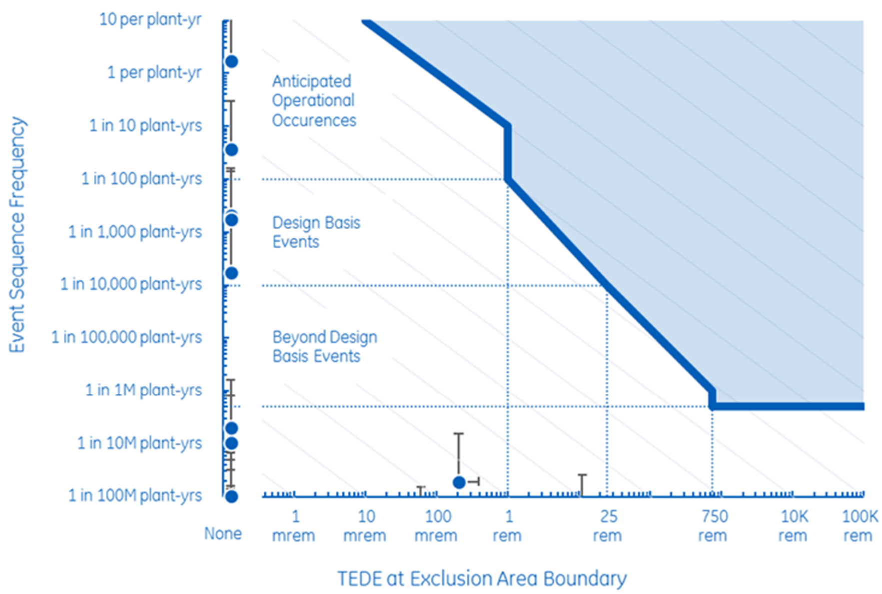

Figure 3-16 Comparison of PRISM LBE Frequencies and Consequences and TLRC Frequency – Dose Criteria

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

~~Table 3-7~~

~~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		
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	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-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		

DBE	Design Basis Events	DBA	Design Basis Accidents
	EM pumps trip, and passive RVACS removes decay 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~~ correlation 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~~ correlation that can be ~~calculated and compared against the risk targets~~used to focus the attention of the designer and those reviewing the design and related operational programs to the most significant events and possible means to address those events. . 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.

  LMP Approach:

  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.

  LMP Approach

  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.

  LMP Approach

  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:

  LMP Approach

  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 5×10^{-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 2×10^{-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.

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