Southern Company

Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors Probabilistic Risk Assessment Approach

Draft Report Revision A
Issued For Collaborative Review

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Date

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Abstract

This report, "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors: Probabilistic Risk Assessment Approach," 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 Nuclear Regulatory Commission (NRC) in developing regulatory guidance for licensing advanced non-LWR plants. In this paper, the LMP is seeking:

- NRC's approval of the proposed technology-inclusive (TI) probabilistic risk assessment (PRA) approach for incorporation into appropriate regulatory guidance for advanced non-LWRs
- Identification of any issues that have the potential to significantly impact the use of risk insights derived from the PRA in the selection and evaluation of Licensing Basis Events (LBEs) and safety classification of systems, structures and components (SSCs)

This paper outlines the approach to develop a PRA for advanced non-LWR plants in support of risk-informed and performance-based (RIPB) applications including:

- Evaluation of design alternatives and incorporation of risk insights into early and continuing development of the design
- Input to the selection of LBEs
- Input to the safety classification of SSCs

Future papers under development as part of the LMP shall address how the PRA is used to support additional risk-informed decisions including:

- Selection of performance requirements for the capabilities and reliabilities of SSCs in the prevention and mitigation of anticipated transients and accidents (The proposed application of special treatment is based on the method of defining risk significance as described in this paper.)
- Risk-informed and performance-based evaluation of defense-in-depth adequacy

Future advanced non-LWR license applications will include a design-specific PRA that is capable of supporting the above listed applications. When introduced at an early stage of the design, the PRA is expected to result in a more efficient risk management process. This paper outlines the relevant regulatory policy and guidance for this type of PRA, describes the approach to be followed for the development of the PRA, and sets forth certain issues for review and discussion in order to facilitate successful design and more safety focused preparation and review of the license application.

Key elements discussed in this paper include the PRA scope and objectives, regulatory guidance used in the formulation of these objectives, and the methodology for factoring the objectives into the proposed TI PRA framework. These PRA elements are first described in terms of a TI framework supplemented with examples of PRA models for specific non-LWR designs including a modular high temperature gas-cooled reactor and a pool-type sodium-cooled fast reactor.

The PRA approach that will be used to support the advanced non-LWR license design is broadly applicable to both single reactor and multi-reactor module plants and is intended to support both design certification and site-specific license applications. As described herein, the PRA is introduced at an early stage in the design, and subsequently upgraded in terms of scope and level of detail at various design and licensing stages as the design matures and the design and siting details are defined. At each stage of the design/PRA development process, information from the PRA will be available to support decisions on the selection and evaluation of design options and to help formulate requirements on the capability and reliability of SSCs in the prevention and mitigation of accidents.

Acknowledgments

The LMP acknowledges the invaluable input received from the B. John Garrick Institute for the Risk Sciences, which was tasked with reviewing the Next Generation Nuclear Plant (NGNP) and LMP draft white papers on PRA development, LBE selection, SSC safety classification and risk-informed evaluation of defense-in-depth. In their review of the PRA papers, the Garrick Institute noted that the referenced examples from the modular high temperature gas-cooled reactor and sodium cooled fast reactor communities already benefit from a long history of PRA development, whereas other advanced non-LWRs are at a much earlier stage of developing a supporting probabilistic and deterministic safety basis. In response to this input, additional guidance has been incorporated into this paper on how to begin the PRA development for a new reactor that may not benefit from a rich PRA history. In addition to this input, John Garrick has provided his personal review comments on an earlier draft of this paper that reflects his vast experience with developing and applying PRA over many decades and covering a wide spectrum of industries. His review includes counsel to avoid delays in addressing external hazards, to embrace the full quantification of uncertainty, and to heed the lessons from the Fukushima Daiichi accident. A summary of the Garrick Institute review is provided in Section 4.3 of this paper.

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List of Abbreviations

ACRS	Advisory Committee on Reactor Safeguards
ACS	Auxiliary Cooling System
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
BDBE*	Beyond Design Basis Event
BOP	balance-of-plant
CDF	core damage frequency
CFR	Code of Federal Regulations
COL	Combined License
DBA	Design Basis Accident
DBE*	Design Basis Event
DID	defense-in-depth
DOE	Department of Energy
EM	electromagnetic
ESD	event sequence diagram
FMEA	failure modes and effects analysis
FR	Federal Register
HPB	helium pressure boundary
HPS	Helium Purification System
HRA	human reliability analysis
HTGR	high temperature gas-cooled reactor
HVAC	heating, ventilation, and air-conditioning
IAEA	International Atomic Energy Agency
IE	initiating event
IHTS	Intermediate Heat Transport System
IHX	Intermediate Heat Exchanger
IPS	Investment Protection System
IRF	inherent reactivity feedback
ISAM	Integrated Safety Assessment Methodology
JCNRM	Joint Committee on Nuclear Risk Management
LBE*	Licensing Basis Event
LERF	large early release frequency
LMP	Licensing Modernization Project
LOCA	loss of coolant accident
LOF	loss of primary forced flow
LOHS	Loss of heat sink
LWR	light water reactor
mHTGR	modular high temperature gas-cooled reactor
MHTGR	a specific mHTGR designed by General Atomics
ML	Main Loop
MLD	Master Logic Diagram
non-LWR	non-light water reactor
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OCS	Operational Plant Control System
PBMR	Pebble Bed Modular Reactor
PHA	process hazard analysis
PIRT	Phenomena Identification and Ranking Table
POS	plant operating state
PRA	probabilistic risk assessment
PRISM	Power Reactor Innovative Small Module
PSA	Probabilistic Safety Assessment
QHO	Quantitative Health Objective
RB	reactor building
RCCS	Reactor Cavity Cooling System
RIPB	risk-informed and performance-based
RPS	Reactor Protection System
RVACS	Reactor Vessel Auxiliary Cooling System
SAP	Safety Assessment Principle
SCS	Shutdown Cooling System
SFR	sodium-cooled fast reactor
SRM	Staff Requirements Memorandum
SSC	structures, systems, and components
SU/SD	Startup/Shutdown
TEDE	Total Effective Dose Equivalent
TI	technology-inclusive
TI-RIPB	technology-inclusive, risk-informed, and performance-based
TLRC	Top Level Regulatory Criteria
ULOF	unprotected loss of primary forced flow
ULOHS	unprotected loss of heat sink
UTOP	unprotected transient overpower
US	United States
USS	Ultimate Shutdown System

*These terms have special meanings defined in this document.

1.0 INTRODUCTION

1.1 Purpose

Many of the current regulatory requirements for United States (U.S.) nuclear power plants are based on light water reactor (LWR) technology used for generation of electricity, necessitating changes to the LWR framework[*] to facilitate efficient, effective, and predictable licensing expectations for a spectrum of novel, advanced, non-light water reactors (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. The LMP objective is to assist the Nuclear Regulatory Commission (NRC) in developing regulatory guidance for licensing advanced non-LWR plants.

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 through issuance of NRC regulatory guidance that supports licensing of advanced non-LWRs. These proposals are intended to lead to 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.

These proposals are technology-inclusive, risk-informed, and performance-based (TI-RIPB). The modernized framework is technology-inclusive (TI) to accommodate the variety of technologies expected to be developed (implementation will inherently 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 (SSCs) to prevent and mitigate accidents. By utilizing a risk-informed, performance-based approach for the Licensing Basis Event (LBE) selection process, the design and licensing efforts are more closely aligned with the safety outcome 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 NRC visions for licensing and deploying advanced non-LWR plants.

The new framework consists of elements including: establishment of TI-RIPB LBE selection, classification of SSCs, and establishment of predictable means to determine and preserve

[*] "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 amended regulatory guidance and implementation proposals (i.e., near-term changes in actual regulation are not anticipated as part of LMP initiatives).

[†] The regulatory framework was defined in SECY-2000-0191, "High-Level Guidelines for Performance-Based Activities" to include the regulation and its supporting regulatory guides, standard review plans, technical specifications, NUREGs, and inspection guidance. It is in this context that the term is generally used in this paper.

adequate defense-in-depth (DID). 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 (PRAs) as part of the early and continuing lifecycle of new designs; and establishment of performance-based licensing criteria in lieu of LWR-centric prescriptive requirements. These elements are supported by reviews of past regulatory precedents and policies to make maximum use of existing approaches and NRC decisions, as well as assessments of current state of the art analytical tools. Gap analyses are used to identify where new or revised requirements are needed for a TI-RIPB framework and propose changes in language or approach to allow the framework changes to be used effectively.

The relationship between the main topics described above is represented in Figure 1-1. A simple diagram cannot capture these relationships comprehensively because the development process for a licensing framework is iterative, not serial. Feedback loops are difficult to represent in a simple figure, and some outputs not shown. Nonetheless, this figure is intended to provide a generalized context for the major activities and how they fit into the overall framework.

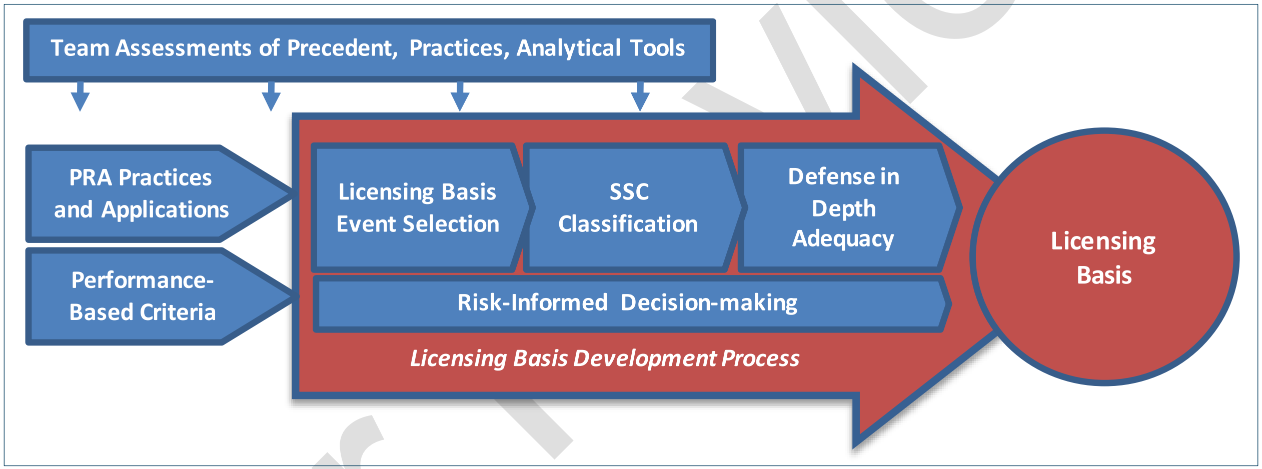

Team Assessments of Precedent, Practices, Analytical Tools
PRA Practices and Applications
Performance-Based Criteria
Licensing Basis Event Selection
SSC Classification
Defense in Depth Adequacy
Risk-Informed Decision-making
Licensing Basis Development Process
Licensing Basis

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

This white paper reviews the relevant regulatory precedents for guidance in the performance of a PRA to support a TI-RIPB design and licensing process. Inputs from the PRA are used in:

1. Supporting and evaluating the development of the design
2. Identifying the spectrum of LBEs to be considered
3. Evaluating the risk significance of LBEs against frequency-consequence evaluation criteria derived from Top Level Regulatory Criteria (TLRC)
4. Performing an integrated risk assessment of advanced non-LWR plants that may be comprised of two or more reactor modules and associated non-core sources of radioactive material
5. Safety classification of SSCs

6. Development of performance criteria for the reliability and capability of SSCs in the prevention and mitigation of accidents
7. Determining integrated plant performance margins compared to TLRC performance-based objectives
8. Exposing and evaluating sources of uncertainty in the identification of LBEs and in the estimation of their frequencies and consequences, and providing key input to the evaluation of the adequacy of DID
9. Providing risk and performance-based insights into the evaluation of the design DID adequacy
10. Supporting other risk-informed and performance-based (RIPB) decisions

The focus of this paper is the development of a PRA that addresses the first five of the above applications associated with design support, identifying and evaluating LBEs, and safety classification of SSCs. Another LMP paper or papers will address the remaining applications of PRA listed above. Each of the above applications is an example of RIPB decision making under the LMP framework.

This paper builds on the development of the PRA white paper for DOE's Next Generation Nuclear Plant (NGNP)[1] and is intended for use with a spectrum of advanced non-LWRs including modular high temperature gas-cooled reactors (mHTGRs), molten salt reactors, sodium-cooled fast reactors, and other non-LWR concepts. The TI PRA approach described herein is intended to support the approach to selection of LBEs and safety classification of SSCs for advanced non-LWR plants described in Reference [2].

1.2 Objective of This Paper

This paper describes a technology-inclusive approach for the development and use of a PRA to support RIPB decisions associated with design and licensing of advanced non-LWR plants. The objectives of this PRA paper are to:

- Identify supporting regulatory guidance, precedents, and available references providing the bases of the proposed PRA approach
- Identify the similarities and differences between the LMP approach to PRA development and use and the approach that has been followed for LWRs
- Identify the key technical issues that will need to be resolved for the successful application of the PRA to advanced non-LWRs
- Describe the approach for using available guides, standards, and peer review processes to assure the technical adequacy of the PRA during design development and licensing
- Define the approach to developing the PRA so that it can be used to provide input to the selection of LBEs, information to select the safety classification of SSCs and associated

safety-related design criteria, the formulation of special treatment requirements, and to perform a risk-informed evaluation of defense-in-depth

- Describe the approach to the PRA treatment of the integrated risk from operation of a multi-reactor module plant*

1.3 Scope

The PRA approach described in this paper applies to a spectrum of advanced non-LWR designs including mHTGRs, molten salt reactors, and sodium-cooled fast reactors and is intended to be reactor technology-inclusive. This white paper discusses the use of the PRA in the selection and classification of LBEs using criteria that focus on acceptable risks to the public health and safety. Risks to the worker will be discussed at a later date, as will security-related events. Worker and security-related risks are not included in the scope of the PRA. Such risks will be addressed using deterministic criteria consistent with operating and advanced LWRs under the proposed LMP framework.

Section 2 of this white paper provides an overview of the regulations and guidance considered during development of the proposed PRA approach. The TI-RIPB approach to PRA is described in Section 3 and builds upon an approach that was developed for the DOE's NGNP.[1] This is accomplished by incorporating insights from NRC and Advisory Committee on Reactor Safeguards (ACRS) reviews of the NGNP approach and by considering PRA applications in a TI manner. This review also considers events and developments in the intervening period following the NGNP work, such as new insights from the Fukushima Accident, and more recent developments in the incorporation of RIPB elements into the regulatory framework.

Section 3 includes a TI approach to performing a PRA with specific examples of PRA models that have been developed for the MHTGR (a specific mHTGR designed by General Atomics) and the Power Reactor Innovative Small Module (PRISM) liquid-metal reactor. In Section 4, key challenges and technical issues to performing the non-LWR PRA are discussed. The approach to achieving technical adequacy in light of these issues is presented. Section 5 summarizes the top priority licensing topics to be discussed with the NRC staff and examines how the proposed approach to PRA meets the existing regulatory foundation for RIPB decision-making.

The PRA methodology described in this white paper is intended for use on advanced non-LWR designs, and it is intended to be applied at various discrete points along the entire reactor design-operation life cycle. It is intended that the PRA be introduced at an early stage of design and noted that the scope and level of detail of the PRA will be consistent with the level of detail of the evolving design and site characteristics.

* The term "plant," as it is used in this document means a nuclear plant that may or may not employ a modular design. A "modular design" indicates 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.[3]

1.4 Summary of Outcome Objectives

The LMP objective is to assist the NRC to develop regulatory guidance for licensing advanced non-LWR plants. In this paper, the LMP is seeking:

- NRC's approval of the proposed TI PRA approach for incorporation into appropriate regulatory guidance for advanced non-LWRs
- Identification of any issues that have the potential to significantly impact the use of risk insights derived from the PRA in the selection and evaluation of LBEs and safety classification of SSCs

Outcome objectives for additional uses of the PRA in RIPB decisions beyond LBE selection and SSC safety classification will be addressed in future LMP papers.

The LMP project is seeking agreement on the following specific statements regarding the PRA approach:

- The scope and technical approach for advanced non-LWR PRAs outlined in this paper are appropriate for the intended applications of the PRA in the design, construction, and operating license application for advanced non-LWR plants including mHTGRs, molten salt reactors, sodium reactors, and other advanced non-LWR concepts. These PRA applications include input to:
    - Evaluation of design alternatives and incorporation of risk insights into the design
    - Selection of LBEs* including the Design Basis Accidents (DBAs)
    - SSC safety classification and special treatment requirements
    - Selection of performance-based targets for the reliability and capability of SSCs within the scope of the PRA
    - RIPB evaluation of DID adequacy
- The road-map presented in this paper for introducing the PRA at an early stage in the design and progressively increasing the scope and level of detail of the PRA models and documentation consistent with the scope and level of detail of the supporting design and siting characterization is appropriate. The iterative nature of the PRA and design development creates a need to review and revise the supported RIPB decisions in order to incorporate new risk insights.
- The TI approaches to initiating event (IE) selection, event sequence development, end-state definition, definition of risk metrics, definition of risk importance measures, and risk-

* As explained more fully in the Licensing Basis Event white paper, LBEs include all the events considered as part of the design and licensing basis including Anticipated Operational Occurrences (AOOs), Design Basis Events (DBEs), Beyond Design Basis Events (BDBEs) and Design Basis Accidents. AOOs, DBEs, and BDBEs are derived from the PRA results and DBAs are deterministically derived to conservatively bound the events within the design basis.

significance determination outlined in this paper are technically adequate for the intended PRA applications.

- The TI approaches to the treatment of inherent characteristics and passive SSCs outlined in this paper are technically adequate.
- The TI approach to using deterministic engineering analyses* for assessing the plant response to IEs and event sequences, success criteria, and mechanistic source terms is appropriate for the proposed risk-informed advanced non-LWR design and licensing approach.
- The TI approach to the development of PRA data outlined in this paper, including the use of applicable data from non-nuclear sources, LWRs, expert opinion, and treatment of uncertainty, is a technically adequate approach for the advanced non-LWR PRA.
- The TI process for PRA treatment of uncertainties in the estimation of accident frequencies and the quantification of mechanistic source terms and consequences in the PRA is a technically adequate approach for the purpose of developing and analyzing the results of the PRA.
- The TI approach for the PRA treatment of multi-unit or multi-module plants including the delineation of accidents involving single and multiple reactor modules and radiological sources is technically adequate to support licensing of single and multi-module plant configurations. It is recognized that case studies in the application of PRA to non-core source of radioactive material are lacking.
- The approach to establishing the technical adequacy of the PRA for its intended RIPB applications based on the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard for Advanced non-LWR Plants and supporting peer reviews as described in this standard is acceptable. It is recommended that NRC take an active role in contributing to, reviewing, and endorsing this standard when the current trial use period for that standard is completed and the American National Standards Institute (ANSI) version of this standard is developed.

1.5 Relationship to Other NGNP Topics/Papers

The LMP approach to PRA has significant interrelationships to other topics being investigated within the scope of the LMP as described below.

Licensing Basis Event Selection Approach

Key inputs to the selection of LBEs are derived from a PRA evaluation of the advanced non-LWR plant. These inputs together with deterministic inputs are used as part of a TI-RIPB approach for the selection and evaluation of LBEs.[2]

* Deterministic engineering analyses referred to here include reactor physics, thermo-fluid analyses, structural analyses, etc. that are necessary to predict the plant response to events, success criteria development, analysis of physical processes and phenomena to resolve the event sequence end states and develop mechanistic source terms.

SSC Safety Classification and Performance Requirements Approach

Information developed from and used in the PRA to define event sequences and evaluate their frequencies and consequences is an input to the SSC safety classification and development of SSC performance requirements. Information from the PRA is used to establish the necessary and sufficient conditions of SSC capability and reliability in order for LBE frequencies, consequences, and uncertainties to stay within the frequency-consequence evaluation criteria derived from the TLRC and to implement risk management strategies to control the total integrated risk of the plant. Reliability requirements for SSCs are determined based on the need to maintain each LBE within its LBE category (Anticipated Operational Occurrence, Design Basis Event, or Beyond Design Basis Event). RIPB SSC capability requirements are defined in part by the selected design margins between the LBE frequencies and dose limits for that LBE category. Special treatment requirements for SSCs are derived to achieve the necessary and sufficient degree of reliability and capability of the SSCs. This will be discussed in a companion white paper on the LMP SSC safety classification approach.

Defense-in-Depth Adequacy

The PRA models and supporting assumptions are based in part on the plant capabilities for DID reflected in the design, as well as assumptions about the limits placed on design and operation of the plant by assumed programmatic DID measures. Information developed in the PRA is used to help evaluate the SSCs responsible for preventing and mitigating accidents. The PRA also plays an important role in the identification of key sources of uncertainty, and this supports a feedback loop to identify possible enhancements to plant capability and programmatic aspects of DID. Hence, the PRA provides important input to the risk-informed evaluation of DID, complements the NRC’s deterministic approach and traditional DID philosophy, and provides a more objective, RIPB means to systematically demonstrate DID adequacy and preservation. This will be discussed in a companion white paper on the LMP approach to DID adequacy.

2.0 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 support RIPB processes and uses. NRC and international regulations, policies, guidance, and other precedents that are relevant to the use of PRA to support RIPB decisions were reviewed. NRC and ACRS feedback on previous efforts to define a RIPB design and licensing approach for NGNP are also reviewed for guidance. Insights from use of PRA to support the design and licensing of advanced LWRs as well as NRC pre-licensing reviews of advanced non-LWRs are included.

This regulatory precedent review builds on the regulatory review in the NGNP uses of PRA[1] by incorporating more recent developments and precedents and by considering the need to have a reactor technology-inclusive approach for performing a PRA rather than one focused on the specific high temperature gas-cooled reactor (HTGR) technology.

A summary of the documents reviewed for regulatory guidance and insights from relevant precedents is provided in Table 2-1. The regulatory documents include the U.S. Code of Federal Regulations (CFR), 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, Pebble Bed Modular Reactor (PBMR), and NGNP projects, as well as the NRC staff and ACRS reviews and feedback on those initiatives. International perspectives were incorporated into the review based on relevant documents from the International Atomic Energy Agency (IAEA) and the regulatory authority in the United Kingdom.

Table 2-1. Documents Reviewed for Regulatory Bases and Precedents

Category	Document	Applicable Content
NRC Regulations	10 CFR 50.71(h)	PRA requirements for Combined License (COL) applications
	10 CFR 52	PRA requirements for Design Certification Application
	10 CFR 52.1	License terms definitions
NRC Policies	73 Federal Register (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	SECY/SRM 2003-0047	Policy issues related to non-LWR licensing
	SECY 2005-0006	Regulatory structure and policy issues for new plant licensing
	SECY/SRM 2006-0007	Advanced notice of Proposed Rulemaking for TI-RIPB process for advanced reactors
	SECY/SRM 2007-0101	Decision to defer rulemaking until new applicant
	SECY/SRM 2011-0089	NRC Level 3 PRA project status
	SECY/SRM 2015-0168	Disposition of NUREG-2150 Risk Management Task Force recommendations

NRC Guidance	Reg. Guide 1.206	COL requirements for Chapter 19 and PRA
	NUREG-0800, Chapter 19	PRA evaluation review guidance
	Reg. Guide 1.174	Use of PRA in risk-informed decisions approach
	Reg. Guide 1.200	Technical adequacy of PRA
	NUREG-1860	RIPB regulatory structure feasibility study
	NUREG-2150	Proposed risk management regulatory framework
	Near Term Task Force 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 Quantitative Health Objectives
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
	ACRS Letter May 15, 2013	ACRS views on NGNP proposed licensing approach
	NRC Letter June 20, 2013	NRC staff response to May 15, 2013 ACRS letter
	NRC Letter July 17, 2014	NRC report to DOE on NRC staff assessment of NGNP white papers
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
	GE-Hitachi 2017 report	Development and modernization of PRISM PRA
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 TECDOC-626	Safety-related terms for advanced reactors
	IAEA TECDOC-1804	Nuclear safety design requirements
	IAEA SRS-04	Multi-unit PRA technical approach
	United Kingdom Safety Assessment Principals (SAPs)	United Kingdom SAPs
	Canadian Nuclear Safety Commission Multi-reactor unit PRA Workshop	Technical issues in multi-unit PRA
	GIF/RWSG/ISAM Report	Integrated Safety Assessment Methodology for Generation IV Nuclear Systems

3.0 PRA APPROACH FOR ADVANCED NON-LWRS

It is well known in the field of PRA that the risks associated with reactor accidents, as assessed in a PRA, are highly design, plant, and site specific. This is true for any type of reactor, but the range of variabilities in assessed risks is much larger when advanced non-LWR reactor concepts are concerned. The use of different materials for the reactor fuel, moderator, and coolant, and the different safety design approaches for the deployment of radionuclide barriers create fundamental differences in the physical processes and plant responses associated with reactor transients and accidents when compared with an LWR. These differences are reflected in the definition of event sequences, end states, and risk metrics that provide the framework for the advanced non-LWR PRA model within the LMP framework.

Despite these differences among the reactor technologies, the PRA approach that has been successfully applied in the U.S. for risk-informed applications of LWR plants and for meeting PRA requirements for licensing advanced LWRs is fundamentally the same approach that is used for advanced non-LWR technologies. Although the predominant use of PRA in U.S. regulatory activities has been with operating and evolutionary LWR designs, there is a long history of PRA development for non-LWR concepts such as HTGRs and sodium-cooled fast reactors (SFRs). Another distinction to make between LWR and advanced non-LWR PRAs is that the former have been introduced after the plants were designed and licensed, limiting the risk-informed applications to incremental changes to plants that were already built and operated. By contrast, advanced non-LWRs have been primarily used as tool to support the design and to formulate the safety design approaches. Early introduction of the PRA greatly expands the range of risk-informed decisions to include the design itself.

The fact that risk is plant, site, and design specific is a primary justification for using information from a plant, site, and design specific PRA to inform decisions that may impact the level of safety of operating a nuclear power plant. This conclusion is amplified when considering advanced non-LWR technologies. Use of generic models and approaches to inform safety decisions, especially those formulated for LWRs, fail to capture the design and technology specific safety issues associated with the advanced non-LWRs. Hence, before the safety and licensing decisions can be effectively made for advanced non-LWRs, it is imperative that design and technology specific risk insights are developed by performing a plant, site, and design specific PRA.

The purpose of this section is to describe the PRA approach that is envisioned for advanced non-LWRs within the LMP framework. This includes the approach to establish the fit-for-purpose technical adequacy of the PRA for RIPB decisions within the LMP framework. The PRA approach is described in terms of a technology-inclusive approach for building a non-LWR PRA model, a roadmap for expanding the scope and level of detail of the PRA as the design matures, and use of consensus standards for advanced non-LWRs. To demonstrate the capabilities of the approach, two example PRA models are described: one for a modular HTGR, the General Atomics MHTGR, and another for a modular pool-type SFR, the PRISM liquid-metal reactor developed by GE-Hitachi. These examples were selected because they have been supported by mature PRAs, exhibit significant differences between them and are both fundamentally different than an LWR.

The focus of this section is to describe the TI approach to performing a PRA within the LMP framework with a view towards the early applications of the PRA, namely incorporating risk insights into the design, providing input to the selection of LBEs early in the design, and for supporting design decisions on the safety classification of SSCs. Future papers in the LMP will provide additional information on the use of the PRA for other RIPB decisions including formulation of SSC performance requirements and risk-informed evaluation of DID.

3.1 Overview of PRA

The advanced non-LWR PRA provides a logical and structured method to guide the design and evaluate its safety characteristics. This is accomplished by systematically enumerating a sufficiently complete set of reactor design specific event sequences and assessing the frequencies, and consequences of those sequences individually and in the aggregate to identify challenges to the plant's safety functions and to quantify the overall risk profile. As discussed more fully in a companion white paper,[2] the PRA is selected as a tool to help identify the LBEs, in part because of its structured process of identifying event sequences and its ability to account for the dependencies and interactions among SSCs, operators, and the internal and external plant hazards that may perturb the operation of the plant and potentially lead to an accidental release of radioactive material. It is the only approach currently known that has the capability to define the reactor specific event sequences in a systematic and exhaustive manner and using methods supported by industry standards.

Rather than limit the quantification to point estimates of selected risk metrics, the PRA will be structured to give emphasis to the treatment of uncertainties. The quantification of both frequencies and consequences of event sequences and sequence families address uncertainties through the performance of quantitative uncertainty analysis where information is available to perform this function and sensitivity analyses to address other sources of uncertainty that are more difficult to quantify. This uncertainty treatment will be used as an input to a risk-informed evaluation of DID as will be discussed in a companion white paper on that topic. The treatment of uncertainties for the advanced non-LWR design will address the available applicable reactor service experience. The quantification of frequencies and consequences of event sequences and the associated quantification of uncertainties will provide an objective means of comparing the likelihood and consequence of different scenarios and of comparing the assessed level of safety against the applicable performance-based requirements. The sources of uncertainty identified in the uncertainty analysis will be given visibility for deterministic treatment in the selection of LBEs and in the development of principal design criteria.

The PRA will be structured to be able to examine the risk significance of design features and SSCs in the performance of safety functions as called for in the NRC Advanced Reactor Policy Statement.[24]

3.2 Rationale for Use of PRA

PRA is selected as an analysis tool because of its capabilities to:

- Provide a systematic identification and enumeration of design-specific plant operating states, hazard groups, IEs, and event sequences
- Provide a basis for the quantification of risk to public health and safety, and serve as an appropriate and acceptable input to optimization of the design, the selection of LBEs, SSC safety classification, and risk-informed evaluation of DID
- Provide a reasonable and acceptable degree of completeness in the enumeration of reactor technology and design specific event sequences, and the treatment of appropriate combinations of failure modes beyond prescriptive single failure assumptions, including consideration of the potential for multiple failures necessary to determine risk levels, identify LBEs, and perform safety classification of SSCs
- Provide a systematic examination of dependencies and interactions and the role that SSCs and operator actions play in the development of each event sequence and accident scenario
- Provide the capability to display the cause and effect relationships between the plant characteristics and the resulting risk levels that are sufficient to support the identification of LBEs and the safety classification of SSCs and their associated performance-based requirements
- Assess the integrated risks for advanced non-LWRs including event sequences involving two or more reactor modules or radionuclide sources when the design employs a modular reactor concept
- Provide quantitative estimates of accident frequencies and consequences under a realistic set of assumptions with a full quantitative treatment of uncertainties that is supported by available data, expert opinion, and other objective evidence
- Define an appropriate set of technology-inclusive and reactor-specific risk metrics that have the capability to define the significant contributions to risk and provide information to demonstrate DID adequacy
- Apply risk-metrics to the evaluation of potential remedial plant changes or programmatic actions as part of risk-informed decision-making process
- Identify the sources of uncertainty for use in the implementation of DID evaluations and resulting risk management strategies and quantify the impacts of uncertainties on the risk results
- Determine the cause and effect relationships between elements of the safety design approach and the risk profile, including the risk significance of SSCs and design features to support the selection of LBEs and perform safety classification of SSCs
- Provide insights into the provision of special treatments of SSCs commensurate with their safety significance in any given event sequence

- Demonstrate compliance with applicable NRC regulations, guidance, and standards associated with plant safety objectives as well as the performance of PRA for an advanced non-LWR license application

Key assumptions that are used to develop success criteria, to develop and apply probability and consequence models, and to select elements for incorporation into the models will be clearly documented. Assumptions that are made in lieu of as-built and as-procured characteristics for the advanced non-LWR design will also be identified and documented.

3.3 PRA Objectives

The objectives of the PRA are to:

- Provide risk insights into the design of the advanced non-LWR, including the design of SSCs that perform safety functions[*] responsible for the prevention and mitigation of accidents
- Provide an acceptably complete set of event sequences from which to select the LBEs for early introduction into the design and subsequently in the license application
- Assess the integrated risks for advanced non-LWRs including event sequences involving two or more reactor modules or radionuclide sources when the design employs a modular reactor concept
- Provide information needed to effectively manage the risks of multi-module and multi-radionuclide source event sequences to ensure such sequences are not risk significant
- Confirm that the applicable requirements, including the safety goal Quantitative Health Objectives (QHOs) for individual and societal risks, are capable of being met at the site selected for the license application
- Provide input for the development of reactor-specific principal design criteria for the plant
- Support the determination of safety classification, safety-related design criteria and special treatment requirements of SSCs
- Support the identification of emergency planning specifications, including the location of the site boundary as well as the goal of appropriately sizing the emergency planning zones
- Support the development of technical specifications

[*] The term "safety function" as used in this report broadly refers to any function by any SSC that is responsible for preventing or mitigating a release of radioactive material from any radioactive material source within the plant. This includes functions performed by SSCs classified as "safety-related" which are credited during design basis accidents and those performed by any other SSC that is modeled in the PRA. Since the PRA is performed initially prior to the safety classification of SSCs, it is not known *a priori* which modeled SSCs will be considered safety-related when the PRA is initially developed. Hence PRA modeled safety functions should not be confused with the safety classification made in the licensing context.

- Provide insight on the role of advanced non-LWR SSCs in the prevention and mitigation of event sequences as part of the risk-informed evaluation of DID
- Determine the risk significance of design features and SSCs to the extent needed to support LBE selection and safety classification of SSCs
- Meet applicable codes, guides, and standards that ensure the technical adequacy of the PRA
- Provide PRA maintenance and update process that supports risk-informed decisions at appropriate stages in the design, licensing, commissioning, and operation of the advanced non-LWR facility

The above objectives cover a broad spectrum of expected PRA applications. The focus of the technical approach in this section is on the early applications of design support, LBE selection and SSC safety classification. NRC agreement on the PRA objectives is an important outcome of this paper.

3.4 Scope of Advanced Non-LWR PRA

The advanced non-LWR PRA will provide a primary source of candidate event sequences for the selection of LBEs, be a key input to the safety classification and design of SSCs, and provide information to support a risk-informed evaluation of the plant's DID. In view of these applications, completeness and design specificity in the enumeration of event sequences are viewed as especially important outcomes of the PRA. The emphasis placed on the roles of inherent and passive capabilities in the safety design approach of typical advanced non-LWRs requires a comprehensive set of challenges to the advanced non-LWR inherent features and passive SSCs be included. Such a comprehensive set includes a full spectrum of internal events and external hazards that pose challenges to the inherent and passive capabilities of the plant.

The PRA at the time of the advanced non-LWR license application will include the following aspects of a full-scope PRA:

- The potential sources of release of radioactive material, including the sources in the reactor core, primary coolant system pressure boundary, process systems, and fuel handling and storage systems
- All planned operating and shutdown modes, including plant configurations expected for planned maintenance, tests, and inspections
- A full range of potential causes of IEs, including internal plant hardware failures, human operator and staff errors, internal plant hazards such as internal fires and floods, and external plant hazards such as seismic events, transportation accidents, and any nearby industrial facility accidents
- Event sequences that cover a comprehensive set of combinations of failures and successes of SSCs and operator actions in the performance of advanced non-LWR-specific safety functions (These event sequences will be defined in sufficient detail to characterize

mechanistic source terms and offsite radiological consequences comparable to an LWR Level 3 PRA as defined by NUREG/CR-2300.[41])

Quantification of the frequencies and radiological consequences of each of the significant event sequences modeled in the PRA. This quantification includes mean point estimates and an appropriate quantification of uncertainty in the form of uncertainty probability distributions that account for quantifiable sources of parameter and model uncertainty in the accident frequencies, mechanistic source terms, and offsite radiological consequences. An appropriate set of sensitivity analyses will also be performed to envelope sources of uncertainty that are not quantifiable, as described below.

- For advanced non-LWR plants covered under license applications that are comprised of multiple reactor modules, event sequences that impact reactor modules independently as well as those that impact two or more reactor modules concurrently will be defined. The frequencies will be calculated on a per-plant-year basis, and the consequences will consider the number of reactor modules and sources involved in the definition of the mechanistic source terms.
- To support the development of reactor specific regulatory design criteria, the PRA will have the capability of evaluating the cause and effect relationships between design characteristics and risk and supporting a structured evaluation of sensitivities to examine the risk impact of adding and removing selected design capabilities, and setting and adjusting SSC reliability requirements.
- Future advanced non-LWRs are expected to have multiple reactor modules to be located at the same site, with the potential for sharing of systems and structures among modules. The PRA will account for the integrated risk of multiple modules and multiple radionuclide sources to help identify design and operational strategies to effectively manage the risks in a multi-module facility. The existence of multiple modules increases the site-wide likelihood of scenarios that impact a single module independently, and creates the potential for scenarios that involve multiple modules as well as the potential for a mechanistic source term involving releases from two or more reactors. These modular reactor considerations will impact the scope and level of detail of the PRA.

NRC agreement on the necessary scope and structure of PRA development is an important outcome of this paper.

3.5 Roadmap for PRA Evolution as the Design Matures

When the PRA is initially introduced at an early stage in the design, the PRA scope will be focused on internal events and full power initial conditions and event sequences involving the reactor sources of radioactive material. The scope and level of detail of the PRA models will also be simplified to be in alignment with the state of knowledge regarding the definition of the design, the safety design approach, and systems design concept. As the design matures and more design definition and details become available, the scope of the PRA will be broadened to address other plant conditions and progressively confirm the plant capability to meet safety objectives. The PRA will only achieve a full scope status prior to plant operation when all the design and testing information (most of it confirmatory) is included. However, the PRA at the completion of the conceptual design should be sufficient to identify an appropriate set of LBEs.

During the process, questions raised in the development of the PRA model framed around the fundamental risk analysis question: “What can go wrong?” will be considered by the design team to assist in defining the challenges that need to be considered to complete the design. These challenges are reflected in the PRA approaches for the systematic and exhaustive enumeration of IEs, event sequences, and logic models for identifying the cause of each event. This enables a structured way to address the remaining two risk questions of “How likely is it” and “What are the consequences?’ As such, the scope of the PRA and level of detail of the PRA mature as the design evolves in an iterative manner.

To meet current regulatory requirements for a new plant license, the PRA will include a full treatment of internal and external events and hazards initiated from applicable plant operating states consistent with then applicable NRC endorsed PRA standards. However, beyond meeting these requirements in this application, the PRA is introduced at a sufficiently early stage of the design to enable the designer to identify the expected LBEs that need to be considered to minimize the potential for costly back-fits later. This approach to using the results of the early PRA to inform the LBE selection process is used to make the LBE selection process systematic, reproducible, and sufficient to identify reactor specific and unique safety issues. This is explained more fully in the companion LMP white paper on LBE selection.[2]

Given the fundamental differences in the design and selection of materials with unique inherent safety characteristics among the advanced non-LWRs and between those and the existing LWRs the current approach of having the applicant propose a list of LBEs with the design for NRC to review yields too much uncertainty in planning, scheduling, and budgeting a design and license application. The iterative nature of the design evolution and PRA maturity means that the initial set of LBEs determined at an early stage of design will be refined and updated periodically and finally confirmed before the application is submitted. However, this approach is expected to converge more quickly on a successful design and licensing outcome rather than trying to derive LBEs from the current LWR-centric requirements.

Prior to first introduction of the PRA, it is necessary to develop a technically sound understanding of the potential failure modes of the reactor concept, how the reactor plant would respond to such failure modes, and how protective strategies will be incorporated into formulating the safety design approach. The incorporation of safety analysis methods appropriate to early stages of design, such as process hazard analysis (PHA) tools, provides industry-standardized methods to ensure that such early stage evaluations are systematic, reproducible and as complete as the current stage of design permits. A suitable reference for performing such PHA evaluations is Reference [72]. PHA methods include hazard and operability assessment and failure modes and effects analysis (FMEA) which are recognized by in the ASME/ANS advanced non-LWR PRA standard as systematic and reproducible methods for comprehensive hazard assessment. PHA may be regarded as a precursor to the development of the PRA and is actually part of the PRA methodology.

Additional guidance for developing the initial knowledge base for a PRA model can be found in a report on an Integrated Safety Assessment Methodology (ISAM) developed by the Generation IV Risk and Safety Working Group.[73] The ISAM approach includes the following elements that are illustrated in Figure 3-1, which was reproduced from the report:

- Qualitative Safety Features Review
- Phenomena Identification and Ranking Table
- Objective Provision Tree (a DID evaluation)
- Probabilistic Safety Assessment[*]
- Deterministic and Phenomenological Analysis

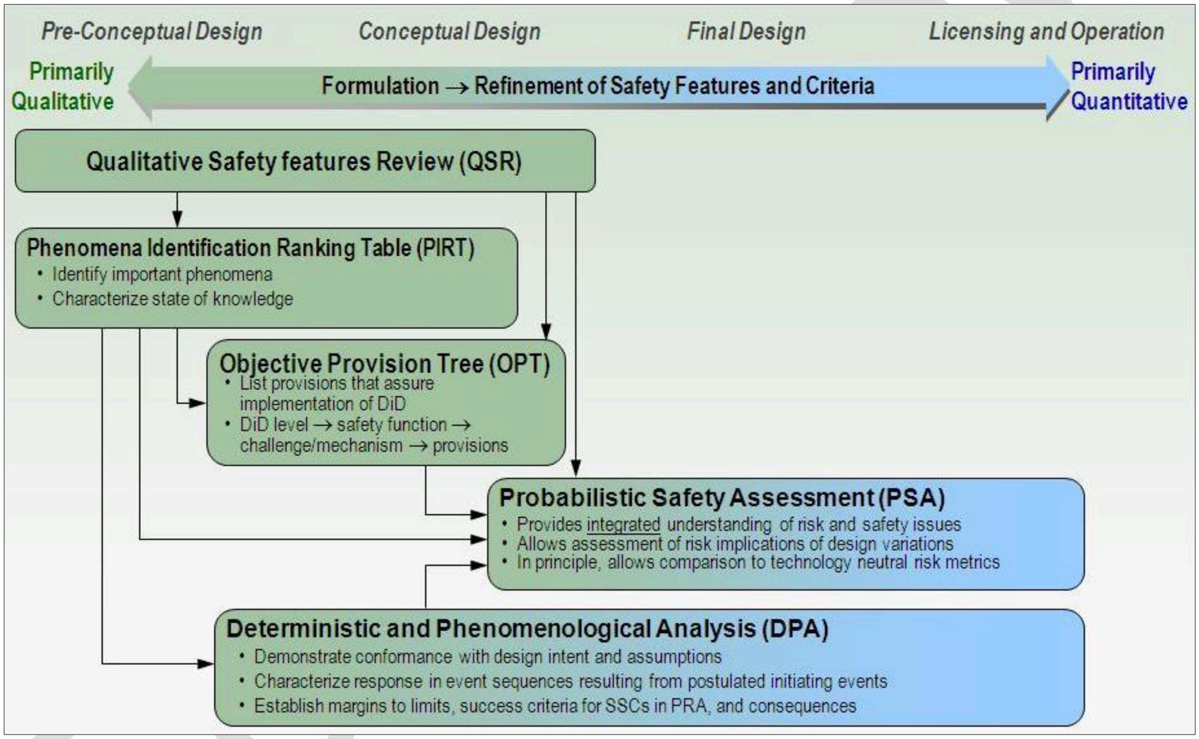

Pre-Conceptual Design
Conceptual Design
Final Design
Licensing and Operation
Primarily Qualitative
Formulation → Refinement of Safety Features and Criteria
Primarily Quantitative
Qualitative Safety features Review (QSR)
Phenomena Identification Ranking Table (PIRT)
• Identify important phenomena
• Characterize state of knowledge
Objective Provision Tree (OPT)
• List provisions that assure implementation of DiD
• DiD level → safety function → challenge/mechanism → provisions
Probabilistic Safety Assessment (PSA)
• Provides integrated understanding of risk and safety issues
• Allows assessment of risk implications of design variations
• In principle, allows comparison to technology neutral risk metrics
Deterministic and Phenomenological Analysis (DPA)
• Demonstrate conformance with design intent and assumptions
• Characterize response in event sequences resulting from postulated initiating events
• Establish margins to limits, success criteria for SSCs in PRA, and consequences

Figure 3-1. ISAM Methodology[73]

The ISAM approach is generally consistent with the approach to PRA advanced in this paper with the following observations:

- The LMP approach recommends an earlier introduction of PRA than does the ISAM approach.
- The Probabilistic Safety Assessment presentation in ISAM retains several LWR-specific concepts, such as the Level 1-2-3 PRA framework, and does not explicitly identify the role

[*] International documents frequently use the term “Probabilistic Safety Assessment,” which has the same meaning as PRA.

of a simplified high level PRA that would be used to guide the conceptual design development.

- ISAM recognizes the need for technology-inclusive risk metrics. However, it attempts to redefine core damage frequency (CDF) in a way that applies to all reactors, which is problematic for many advanced non-LWRs. Even when a core damage state can be defined, there is no basis for applying the acceptance criteria for CDF which have been established for large LWRs.
- All the elements of the ISAM including the Qualitative Safety Features Review, Phenomena Identification and Ranking Table, Objective Provision Tree, and Deterministic and Phenomenological Analysis, as well as the Probabilistic Safety Assessment, are encompassed within the proposed LMP framework as providing critical inputs to a PRA. Their equivalents must be embodied within the deterministic and probabilistic safety analyses for advanced non-LWRs.
- A key strength of ISAM is the incorporation of DID considerations at an early stage of design.

When introduced in the pre-conceptual or early stages of the conceptual design, the initial PRA will be simplified in relation to a full scope PRA that is sufficient to meet applicable PRA standards. Typical simplifications in this early stage include:

- Limitation to internal IEs initiated during full power operation modes
- Representation of all PRA safety functions that protect each radionuclide barrier
- Representation of all known SSCs that support each safety function with no assumptions regarding safety classification
- Use of coarse high level system fault models that reflect known design details
- Simplified treatment of common cause failures and human reliability
- Event sequence quantification using generic data engineering judgment sufficient for order of magnitude estimates and initial LBE determinations
- Plant response to events based on available plant response models
- Source terms based on best available information
- Consequences limited to site boundary dose calculations

Despite these simplifications, the PRA would be capable of defining a reasonably complete set of event sequences and order of magnitude estimates of the frequencies and site boundary doses of those involving a release. Hence the PRA should be sufficient to develop an initial set of LBEs to support the early stages of design. As the PRA is upgraded to conform with later stages of design development, the LBEs will be refined, however the DBAs are expected to be reasonably stable. Between major upgrades and updates of the PRA, it is expected that there would be essentially continuous use of the PRA models to inform design trade studies and to evaluate design alternatives. LBEs associated with internal plant hazards, such as fires and floods, and external events are added as sufficient design and siting information to support these analyses becomes available.

Common sense is used to select appropriate times for PRA upgrades and updates to correspond with key hold points in the development of the design. When the plant general arrangement drawings and cable tray layouts are available, the scope of the IEs can be expanded to include internal fires and floods and other internal hazards. The inclusion of other radionuclide sources within the scope of the PRA can begin when the design features of the supporting systems and structures have been developed. When the site characteristics or site parameter envelope is established PRA models for seismic and other external hazards can be introduced. As operational information becomes available additional modes and states may be added and the treatment of human actions can be advanced. As the capabilities for simulations of plant response to events and mechanistic source terms become available the event sequence models may be refined and the consequence estimates revised. Hence the list of LBEs can be expected to be modified several times prior to the license application. However, the designer will have the benefit of risk insights to guide the design and much better predictability of the LBEs as the design is being developed. The evolution of the design, PRA, and LBE definition for the MHTGR project is illustrated in Figure 3-2.

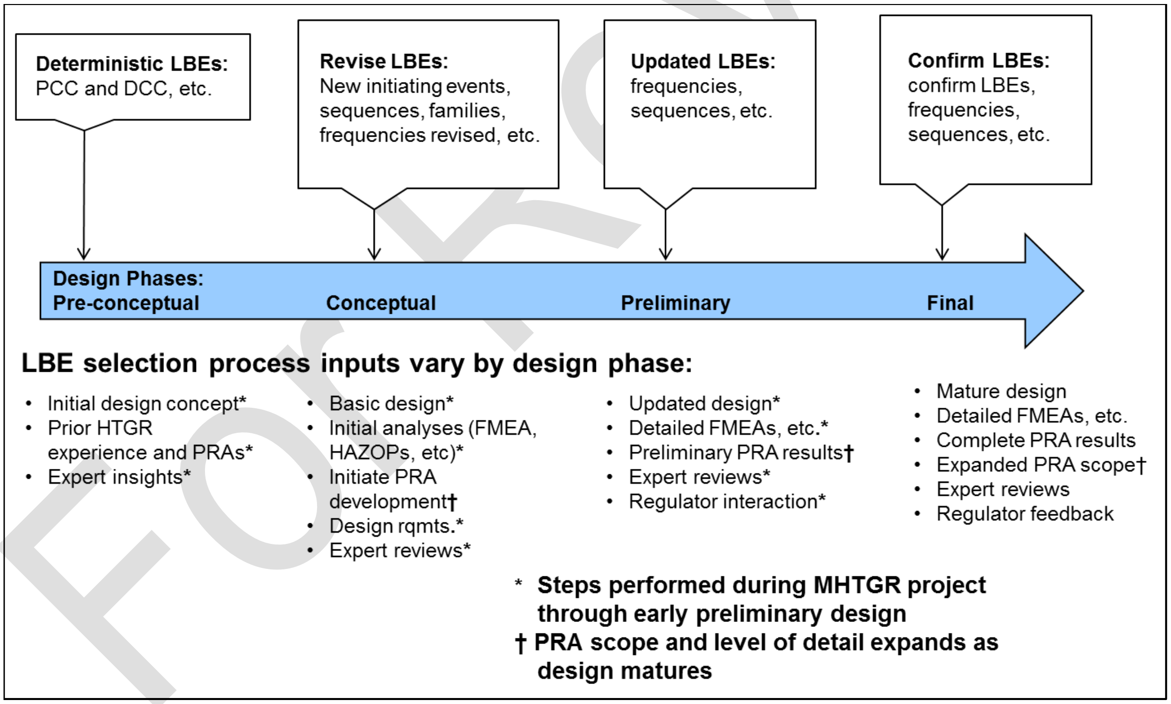

Deterministic LBEs:
PCC and DCC, etc.
Revise LBEs:
New initiating events, sequences, families, frequencies revised, etc.
Updated LBEs:
frequencies, sequences, etc.
Confirm LBEs:
confirm LBEs, frequencies, sequences, etc.
Design Phases:
Pre-conceptual
Conceptual
Preliminary
Final
LBE selection process inputs vary by design phase:
• Initial design concept*
• Prior HTGR experience and PRAs*
• Expert insights*
• Basic design*
• Initial analyses (FMEA, HAZOPs, etc)*
• Initiate PRA development†
• Design rqmts.*
• Expert reviews*
• Updated design*
• Detailed FMEAs, etc.*
• Preliminary PRA results†
• Expert reviews*
• Regulator interaction*
• Mature design
• Detailed FMEAs, etc.
• Complete PRA results
• Expanded PRA scope†
• Expert reviews
• Regulator feedback
* **Steps performed during MHTGR project through early preliminary design**
† **PRA scope and level of detail expands as design matures**

Figure 3-2. Evolution of Design, PRA, and LBE Development for the MHTGR[38]

The initial PRA model is developed in close coordination with the design development and gets input from a number of design analyses that comprise deterministic inputs to the PRA. Several of these key PRA-Design-Analysis interfaces are illustrated in Figure 3-3. When the deterministic inputs are modified as the design evolves, the PRA models are modified as appropriate.

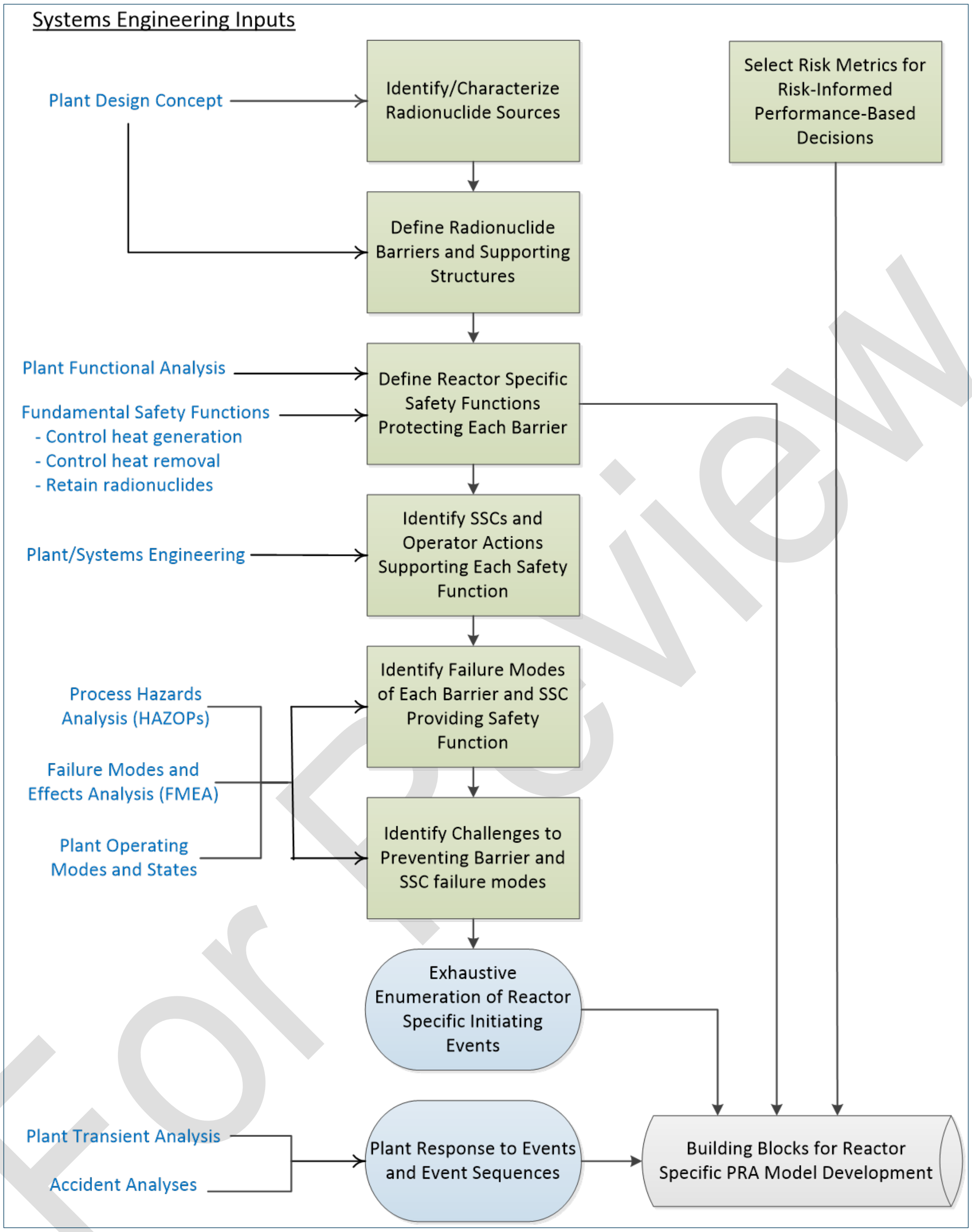

Systems Engineering Inputs
Plant Design Concept
Identify/Characterize Radionuclide Sources
Select Risk Metrics for Risk-Informed Performance-Based Decisions
Define Radionuclide Barriers and Supporting Structures
Plant Functional Analysis
Fundamental Safety Functions
- Control heat generation
- Control heat removal
- Retain radionuclides
Define Reactor Specific Safety Functions Protecting Each Barrier
Plant/Systems Engineering
Identify SSCs and Operator Actions Supporting Each Safety Function
Process Hazards Analysis (HAZOPs)
Failure Modes and Effects Analysis (FMEA)
Plant Operating Modes and States
Identify Failure Modes of Each Barrier and SSC Providing Safety Function
Identify Challenges to Preventing Barrier and SSC failure modes
Exhaustive Enumeration of Reactor Specific Initiating Events
Plant Transient Analysis
Accident Analyses
Plant Response to Events and Event Sequences
Building Blocks for Reactor Specific PRA Model Development

Figure 3-3. Flow Chart for Initial PRA Model Development

The framework for the PRA development is a comprehensive set of design specific IEs and event sequences that are defined by the basic reactor design concept and a number of systems analyses that are performed as a natural part of the reactor design process. The PRA model is systematically developed by defining the sources of radioactive material that are defined by the scope of the PRA, the radionuclide barriers for each source, and the safety functions that protect each barrier. The designer selects and designs the SSCs that provide the barriers and perform the safety functions that protect them.

To develop a robust design that meets all the design requirements including those for operational reliability, availability, maintainability, investment protection and safety, the design is subject to a number of systems analyses including FMEA, PHA, and other analyses specified in the selected design codes and standards. When properly interfaced with the PRA development, these analyses have the capability to provide an exhaustive and systematic search for IEs, which is an early step in PRA model development. The systematic identification of event sequences which are developed with PRA tools such as event sequence diagrams and event trees is accomplished with input from the selection of risk metrics for risk-informed decision making and information from plant transient analysis the designers must perform to design the plant control and protection systems. By integrating the PRA development needs with the early phases of plant design, the additional analyses that need to be performed to support the PRA development may be minimized.

As previously noted, the term "safety function," as used in this report, is any function by any SSC that is responsible for preventing or mitigating a release of radioactive material from any radioactive material source within the plant. Some of these safety functions may later be classified as "required safety functions" if they are necessary or relied upon in the DBA to meet the acceptance criteria for the Chapter 15 safety analysis, or "supportive safety functions" if they are not necessary to meet the Chapter 15 safety analysis criteria but still play a role in accident prevention and mitigation and part of the plant capabilities for DID. Following the development of the initial PRA while selecting the LBEs that will be analyzed as DBAs, those SSCs that perform required safety functions will be classified as "safety-related" and special treatment requirements will be developed to ensure they have the necessary and sufficient capability and reliability to assure the TLRC are satisfied.

Safety functions are defined starting with fundamental reactor inclusive functions of controlling heat generation, controlling heat removal, and retaining radionuclides.[47] These are often refined into reactor technology-specific safety functions that reflect the reactor concept and unique characteristics of the reactors defining the radionuclide barriers. This then leads to the reactor technology specific SSCs that the designer has selected to perform each function as well as to perform non-safety-related functions for energy production, investment protection, and other non-safety requirements.

3.6 Advanced Non-LWR PRA Elements

The advanced non-LWR PRA will be organized into elements that are consistent with the way in which PRA elements have been defined in the ASME/ANS PRA Standard for Advanced Non-LWR Plants.[43] The PRA elements, which may be considered building blocks of the PRA models are listed below. The role these elements play in the development and quantification of the advanced non-LWR event sequence model is illustrated in Figure 3-4.

- Definition of Plant Operating States
- Initiating Events Analysis
- Event Sequence Development
- Success Criteria Development
- Thermal and Fluid Flow Analysis
- Systems Analysis
- Data Analysis
- Human Reliability Analysis
- Internal Flooding Analysis
- Internal Fire Analysis
- Seismic Risk Analysis
- Other External Events Analysis
- Event Sequence Frequency Quantification
- Mechanistic Source Term Analysis
- Radiological Consequence Analysis
- Risk Integration and Interpretation of Results
- Peer Review

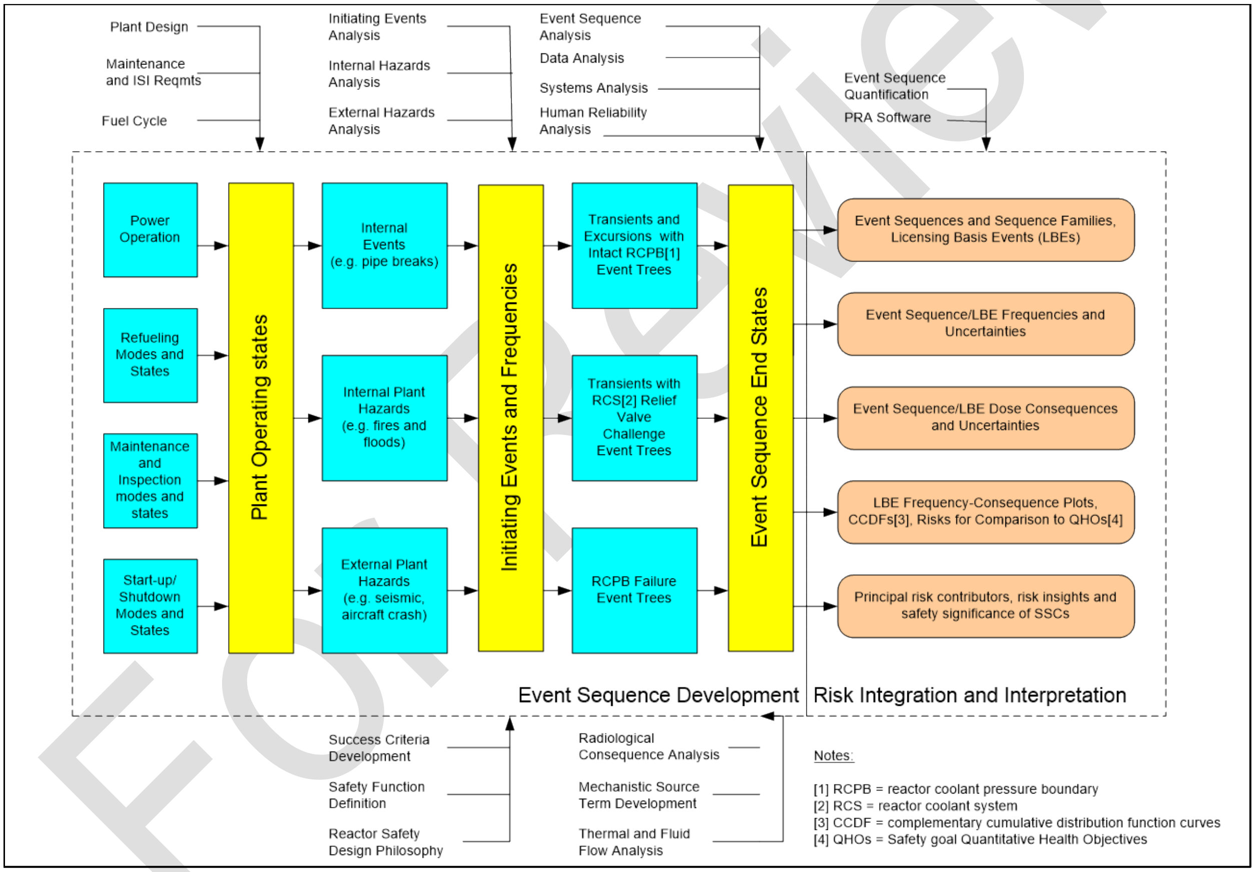

Plant Design
Maintenance and ISI Reqmts
Fuel Cycle
Initiating Events Analysis
Internal Hazards Analysis
External Hazards Analysis
Event Sequence Analysis
Data Analysis
Systems Analysis
Human Reliability Analysis
Event Sequence Quantification
PRA Software
Power Operation
Refueling Modes and States
Maintenance and Inspection modes and states
Start-up/ Shutdown Modes and States
Plant Operating states
Internal Events (e.g. pipe breaks)
Internal Plant Hazards (e.g. fires and floods)
External Plant Hazards (e.g. seismic, aircraft crash)
Initiating Events and Frequencies
Transients and Excursions with Intact RCPB[1] Event Trees
Transients with RCS[2] Relief Valve Challenge Event Trees
RCPB Failure Event Trees
Event Sequence End States
Event Sequences and Sequence Families, Licensing Basis Events (LBEs)
Event Sequence/LBE Frequencies and Uncertainties
Event Sequence/LBE Dose Consequences and Uncertainties
LBE Frequency-Consequence Plots, CCDFs[3], Risks for Comparison to QHOs[4]
Principal risk contributors, risk insights and safety significance of SSCs
Event Sequence Development
Risk Integration and Interpretation
Success Criteria Development
Safety Function Definition
Reactor Safety Design Philosophy
Radiological Consequence Analysis
Mechanistic Source Term Development
Thermal and Fluid Flow Analysis
Notes:
[1] RCPB = reactor coolant pressure boundary
[2] RCS = reactor coolant system
[3] CCDF = complementary cumulative distribution function curves
[4] QHOs = Safety goal Quantitative Health Objectives

Figure 3-4. Overview of Advanced non-LWR PRA Model Elements

These elements are similar to those associated with a full-scope Level 3 PRA for an existing LWR. Some of the key differences expected for the non-LWR PRA are identified below.

- The following design-specific PRA elements are developed specifically for the advanced non-LWR and these are expected to be fundamentally different than those for an LWR:
  - Functional barriers for retention of radioactive material

  - Safety functions
  - SSC available in the design to support each function
  - Success criteria
  - Functional IE categories
  - Plant response to IEs
  - Human actions prior to, in the initiation of, and in response to events modeled in the PRA, including the time frames available for these actions
  - Event sequence end states
  - Mechanistic source terms
  - Radiological consequences
- The event sequences cover relatively frequent events classified as Anticipated Operational Occurrences (AOOs), infrequent events classified as Design Basis Events (DBEs), and rare events classified as Beyond Design Basis Events (BDBEs) based on frequency of occurrence.
- Rather than calculate LWR PRA intermediate risk metrics, such as CDF, the results include calculations of the frequencies and dose consequences of accident families referred to as LBEs. Each LBE is a group of event sequences with similar plant operating state, IE, plant response to performance and failure to perform safety functions, and end-state. The results may also be organized into reactor-specific release category frequencies. Each release category is a grouping of LBEs with similar mechanistic source term.
- For some risk metrics, such as the NRC safety goal QHOs, the risk is aggregated over all the event sequences in the PRA model.
- Event sequence frequencies are calculated on a per-plant-year basis, where a plant may consist of several reactor modules. This facilitates an integrated treatment of risk for an entire multi-module plant. The consequences of event sequences may involve source terms from one, multiple, or all reactor modules or radionuclide sources that comprise the plant. This will facilitate the definition of LBEs for the multi-module design and provide the capability to address the integrated risk of the multi-module plant. This capability of the PRA is necessary to provide risk insights to the designers to enable the effective risk management of multi-module or multi-source accidents.
- The PRA model elements illustrated in Figure 3-4 focus on the reactor scope of the PRA. However, the PRA is also expected to include non-reactor sources of radioactive material.

3.7 Selection of Risk Metrics for PRA Model Development

3.7.1 Overall Plant Risk Metrics

The advanced non-LWR PRA model may be structured differently than the traditional Level 1-2-3 model for an LWR PRA (as defined in NUREG/CR-2300[41]), given the fact that modeled plant damage states may not involve an equivalent to the core damage state that separates the Level 1 and Level 2 parts of an LWR PRA model. Indeed, some advanced non-LWR designs have design features that preclude the type of core damage states that have been defined for an LWR using inherent and passive design features.

Because of the use of different materials for the fuel, moderator, and coolant, LWR risk metrics such as core damage frequency are not useful or relevant for many advanced non-LWR designs. Even in cases where a core damage state may be defined for a non-LWR, its meaning and risk context may be fundamentally different than that for an LWR. This issue was recognized in the development of the ASME/ANS PRA Standard for Advanced non-LWRs,[43] which has adopted TI risk metrics to define PRA requirements for technical adequacy. These TI risk metrics are described below.

- Frequencies of event sequences individually and grouped into accident families having the same or similar plant response and offsite radiological consequences. Accident families may be defined in terms of release categories as the term is used in LWR Level 2 PRAs or into specific LBEs defined by similarity of IE, plant response, mechanistic source term, and offsite radionuclide consequences. Consequences are quantified in terms of offsite early and latent health effects and/or site boundary doses.
- Integrated risks of a given consequence metric, e.g., site boundary dose, number of early or latent health effects, etc. many be calculated by summing the product of the frequency and consequence of each LBE over the full set of LBEs.
- Integrated risks of individual fatalities as needed for comparison to the QHOs.
- Cumulative frequency of exceeding consequences such as large release, early or latent health effects, or a specific site boundary dose.

In addition to the above TI metrics, reactor specific risk metrics defined by the user may be used to define the parameters of the PRA model (for example, frequency of sodium boiling in liquid metal-cooled reactors, frequency of exceeding a fuel damage limit, frequency of pressurized loss of forced cooling in a gas-cooled reactor, etc.) There are requirements for the definition and use of these reactor specific metrics in the advanced non-LWR PRA standard.[43]

Another factor that needs to be considered in the selection of PRA risk metrics is the need to address accident sequences that may involve two more reactor modules or radionuclide sources. The traditional LWR PRA metrics have been used almost exclusively to support PRAs on a one reactor at-a-time basis. This is addressed in the LMP approach to PRA using the following approaches:

- The IEs and event sequences in the PRA will delineate events involving each reactor and radionuclide source separately as well as events involving two or more reactors or sources.
- Dependencies associated with shared systems and structures are explicitly modeled in an integrated fashion to support an integrated risk assessment of the entire plant where the plant may be comprised of two or more reactor modules and non-core radionuclide sources.

- Treatment of human actions will consider the unique performance shaping factors associated with multi-reactor and multi-source event sequences.
- Treatment of common cause failures will delineate those that may impact multiple reactor modules.
- The frequency basis of the event sequence quantification is events per (multi-module/multi-source) plant-year.

A summary of the technical issues that need to be addressed for PRAs involving multiple reactor modules or radionuclide sources is found in Reference [53].

3.7.2 Risk Metrics for Selection of Licensing Basis Events

As discussed more fully in the companion white paper on the LMP approach to LBE selection,[2] there are two types of risk significance evaluations to be performed for the selection and evaluation of LBEs. The first type is an evaluation of the frequencies and consequence of each LBE, expressed in the form of mean values and uncertainty percentiles (5th and 95th percentiles), against frequency-consequence evaluation criteria such as that defined in Figure 3-5 for the LMP framework.

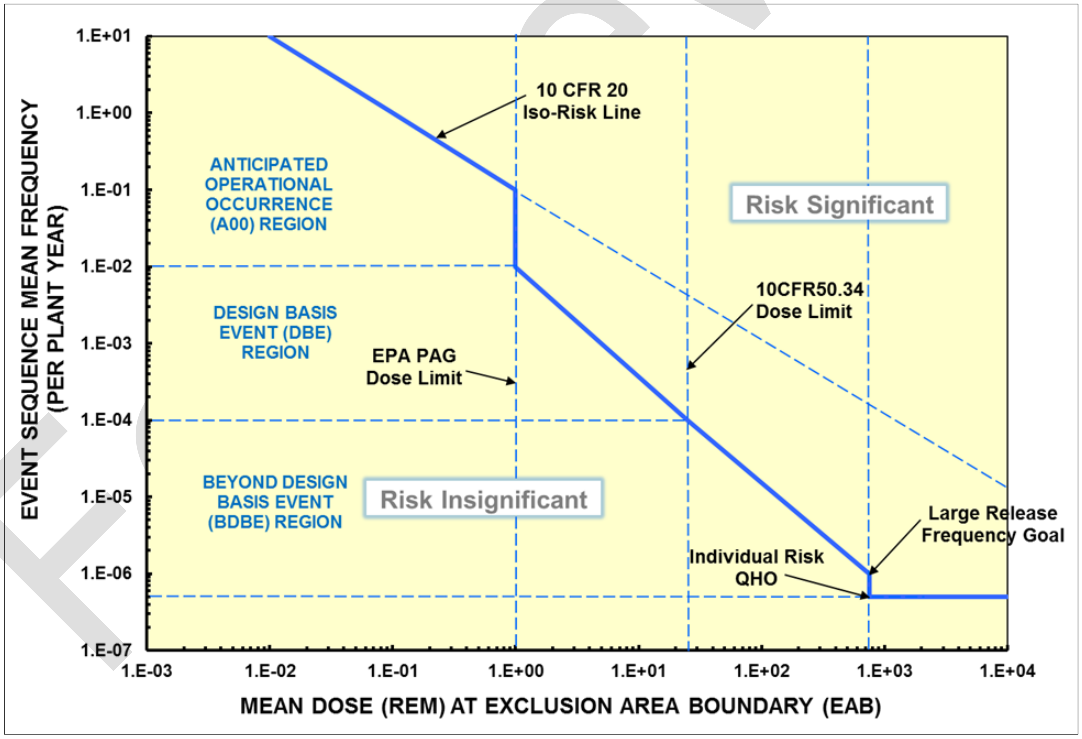


Figure 3-5. Frequency-Consequence Evaluation Criteria for LBEs Proposed for LMP

Each LBE in this evaluation is defined as a family of event sequences modeled in the PRA that groups the individual modeled PRA event sequences according to the similarity of the following elements of the event sequence:

- Plant operating state
- IE
- Plant response to the IE and any independent or consequential failures represented in the event sequence including the nature of the challenge to the barriers and SSCs supporting each safety function
- Event sequence end state
- Number or combination of reactor modules and radionuclide sources affected by the sequence
- Mechanistic source term for sequences involving a release

The event sequence frequencies are expressed in terms of events/plant-year where a plant may be comprised of two or more reactor modules and sources of radioactive material.

In addition to evaluation of each individual LBE, and integrated risk evaluation of the entire plant is performed against the following criteria that have been selected for the LMP project. For this evaluation, the integrated risk of the entire plant is evaluated against four evaluation criteria:

- 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[25] 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.

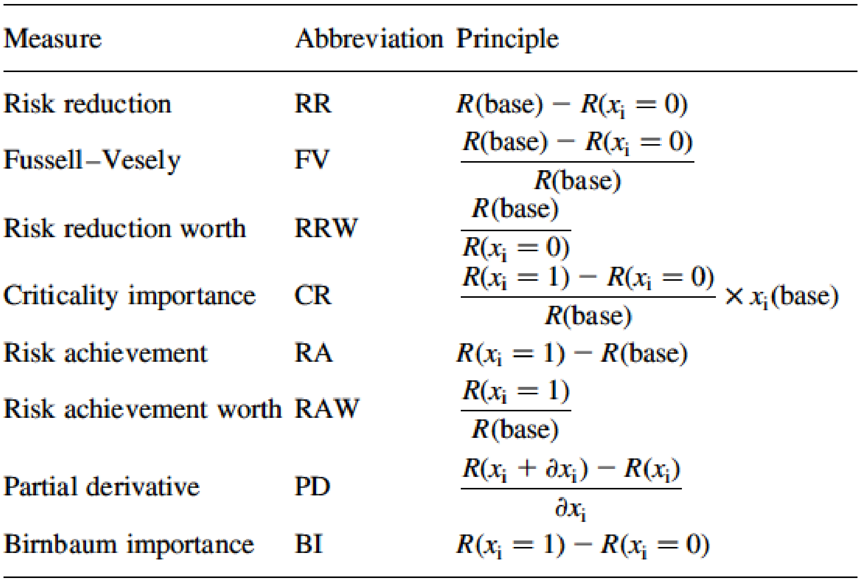
To provide input to the selection of emergency planning zones, the frequency of exceeding the Environmental Protection Agency protective action guideline dose limits would be included in the calculated risk metrics.

3.7.3 Contributors to Risk and Risk Importance Measures

To derive useful risk insights from the results of a PRA, it is necessary to understand the principal contributors to each evaluated risk metric. This is normally achieved by rank ordering the PRA event sequences and sequence minimal cut-sets to identify their relative and absolute contribution to each risk metric and to calculate the risk importance measures that evaluate contributions to basic events that may be common to two or more sequences or cut-sets. For any of the integrated risk metrics, such as the QHOs, the relative risk significance of any LBE may be calculated as a percentage of the LBE risk (product of the LBE frequency and LBE consequence) to the aggregated risk of all the modeled LBEs.

In order to evaluate the risk contributions from basic events that may appear in two or more event sequences or cut-sets, risk importance measures are often used. The most commonly used risk importance measures in PRA are listed in Table 3-1, which is developed from Reference [55]. In this table, the term *R* represents the total risk, *R(base)*, is the risk with each basic event probability set to its base value, and the term x_i represents the probability of a basic event *i*, which may be, for example, the event that a specific valve fails to perform its function.

Table 3-1. Risk Importance Measures[55]



Measure	Abbreviation	Principle
Risk reduction	RR	$R(\text{base}) - R(x_\text{i} = 0)$
Fussell–Vesely	FV	$\dfrac{R(\text{base}) - R(x_\text{i} = 0)}{R(\text{base})}$
Risk reduction worth	RRW	$\dfrac{R(\text{base})}{R(x_\text{i} = 0)}$
Criticality importance	CR	$\dfrac{R(x_\text{i} = 1) - R(x_\text{i} = 0)}{R(\text{base})} \times x_\text{i}(\text{base})$
Risk achievement	RA	$R(x_\text{i} = 1) - R(\text{base})$
Risk achievement worth	RAW	$\dfrac{R(x_\text{i} = 1)}{R(\text{base})}$
Partial derivative	PD	$\dfrac{R(x_\text{i} + \partial x_\text{i}) - R(x_\text{i})}{\partial x_\text{i}}$
Birnbaum importance	BI	$R(x_\text{i} = 1) - R(x_\text{i} = 0)$

In LWR PRAs, the risk metrics used for *R* are typically limited to CDF and large early release frequency (LERF). However, the associated Table 3-1 risk importance measures definitions can also be used with any of the technology-inclusive risk metrics selected for the advanced non-LWR PRA under the LMP framework. These include:

- Frequency of a specific LBE
- Total risk (sum of the product of frequency and site boundary dose) of all the PRA modeled sequences, or individual risk of fatality in the plant vicinity
- Frequency of exceeding a specified site boundary dose
- Individual risk of prompt or latent fatality for comparison to NRC safety goal QHOs

The traditional approach to evaluating risk importance produces only the relative importance of each basic event because the formulas are normalized against the total calculated risk for the plant, *R(base)*. However, this total risk may be very small, especially for advanced non-LWR designs. Indeed, PRAs for evolutionary LWRs have produced estimates of CDF and LERF that are as much as several orders of magnitude lower than those estimated for operating plants. For advanced non-LWR plants, the frequencies of accidents involving a release of radioactive material may be very small and even those accidents with releases may involve very small source terms compared with releases from an LWR core damage accident. Hence, it is appropriate to evaluate risk significance not only on a relative but also on an absolute basis. For this purpose, the risks can be compared against the risk goals rather than the baseline risks. This topic is under discussion for the next edition of the Advanced Non-LWR PRA Standard.

3.8 Example PRA Development for Modular HTGRs

This section summarizes how the technology-inclusive approach to developing a PRA model for advanced non-LWRs is implemented for mHTGR designs of the type supported by the ANS Standard 53.1.[46] The objective is not to present an actual PRA but to show examples of PRA model elements for HTGR designs, such as the MHTGR, whose design is described in Reference [38] and whose supporting PRA is documented in Reference [4]. The PRA model elements presented in this section are representative of other modular HTGRs such as the Exelon PBMR and the NGNP HTGR. This example develops PRA model elements for a plant comprised of four reactor modules and includes both single module and multi-module accidents. Accident sequences involving non-core sources of radioactive material are not included in this example.

The key characteristics of these modular HTGR designs include:

- Graphite moderated and helium cooled
- Ceramic coated particle fuel with robust radionuclide retention capabilities over the full range of operating and accident conditions
- Fuel elements arranged into compacts inserted in to graphite blocks or graphite fuel spheres
- Strong negative temperature coefficient of reactivity
- Passive decay heat removal capability in pressurized and depressurized conditions
- Concentric barriers to radionuclide release including the fuel, helium pressure boundary (HPB), and vented reactor building

3.8.1 Systematic Search for HTGR Initiating Events

As noted in Figure 3-3, the PRA model development begins with a systematic search for IEs for which event sequences need to be defined. A logic model for guiding this search is often referred to as a Master Logic Diagram (MLD). The form of the MLD used to guide IE development for the MHTGR is shown in Figure 3-6. The MLD process starts with the identification of the sources of radioactive material, barriers to fission product release, safety functions that protect each barrier, and initial plant operating states.

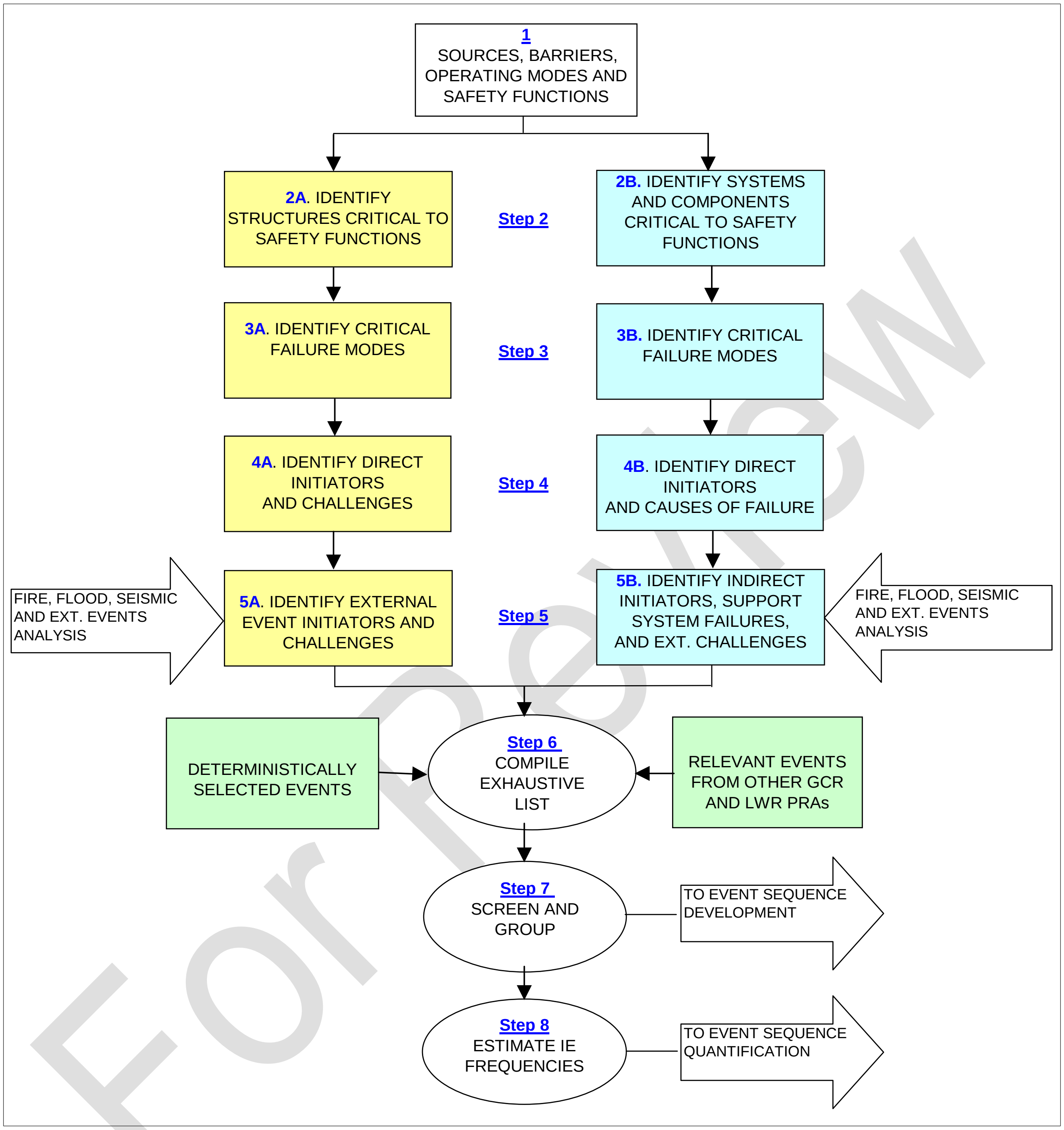

1
SOURCES, BARRIERS, OPERATING MODES AND SAFETY FUNCTIONS
2A. IDENTIFY STRUCTURES CRITICAL TO SAFETY FUNCTIONS
Step 2
2B. IDENTIFY SYSTEMS AND COMPONENTS CRITICAL TO SAFETY FUNCTIONS
3A. IDENTIFY CRITICAL FAILURE MODES
Step 3
3B. IDENTIFY CRITICAL FAILURE MODES
4A. IDENTIFY DIRECT INITIATORS AND CHALLENGES
Step 4
4B. IDENTIFY DIRECT INITIATORS AND CAUSES OF FAILURE
FIRE, FLOOD, SEISMIC AND EXT. EVENTS ANALYSIS
5A. IDENTIFY EXTERNAL EVENT INITIATORS AND CHALLENGES
Step 5
5B. IDENTIFY INDIRECT INITIATORS, SUPPORT SYSTEM FAILURES, AND EXT. CHALLENGES
FIRE, FLOOD, SEISMIC AND EXT. EVENTS ANALYSIS
DETERMINISTICALLY SELECTED EVENTS
Step 6 COMPILE EXHAUSTIVE LIST
RELEVANT EVENTS FROM OTHER GCR AND LWR PRAs
Step 7 SCREEN AND GROUP
TO EVENT SEQUENCE DEVELOPMENT
Step 8 ESTIMATE IE FREQUENCIES
TO EVENT SEQUENCE QUANTIFICATION

Figure 3-6. Master Logic Diagram Guiding the Steps to Selection of MHTGR Initiating Events

The following sources of radioactive material are considered for the MHTGR:

- Sources within the primary system HPB:
  - Fuel elements in core
  - Intact coated particles

  - Failed or defective coated particles
  - Uranium contamination outside coated particles
  - Sources imbedded/attached to graphite components
  - Dust and plateout on HPB surfaces
  - Circulating primary coolant activity
- Sources outside the HPB:
  - Fuel elements in storage systems
  - Helium Purification System (HPS) gas-borne activity
  - Solid and liquid radwaste systems

Table 3-2 summarizes the principal barriers to each of these sources. Once the sources, barriers, and safety functions are defined, the Master Logic Diagram follows a step-by-step process of defining the failure modes of each SSC, the impacts of these modes in challenging the barriers and safety functions, and of identifying direct IEs, as well as challenges posed by internal and external hazards. Two separate paths are followed through these steps in Figure 3-6: one from the viewpoint of each barrier and its set of challenges, and the other from the viewpoint of the SSCs providing safety functions in support of these barriers. The former may be viewed as direct challenges to the integrity of the barriers and the latter as indirect challenges to the barriers.

Table 3-2. MHTGR Radionuclide Sources and Barriers

Radioactive Material Source	Barriers to Radionuclide Transport
Fuel elements in the core	Fuel particle kernel, silicon carbide and pyrocarbon coatings of the fuel particle, fuel matrix and fuel element graphite, HPB (primary circuit), reactor building
Fuel elements outside the core	Fuel particle kernel, silicon carbide and pyrocarbon coatings of the fuel particle, fuel matrix and fuel element graphite, fuel handling and storage systems, reactor building
Non-core sources within the HPB	HPB, reactor building
Other sources within the plant	Various tanks, piping systems and containers, reactor building or ancillary buildings housing waste management equipment

An initial screening is performed for all SSCs in the plant, including the radionuclide transport barriers for SSCs that play no direct or indirect role in supporting a safety function and whose failure does not impact the safety functions of other SSCs or cause an IE are screened out. Failure modes and effects analyses are performed for all unscreened SSCs and radionuclide transport barriers to identify potential internal IEs. An analysis of internal and external plant hazards (including those from co-located facilities) is performed to encompass the remaining challenges to the plant safety functions. These processes ensure that events specific to the MHTGR design are considered. Insights from reviews of nuclear plant operating experience and previous safety and risk analyses are used to ensure completeness of the exhaustive list of events. In the design and licensing of the MHTGR facility, the systematic selection of IEs is viewed as common to both the probabilistic and deterministic elements of the safety analysis approach. This fact is important to understand the way in which deterministic and probabilistic elements have been integrated into the MHTGR design, which is the key advantage of applying PRA technology in the beginning.

3.8.2 MHTGR Safety Functions

The MHTGR PRA includes a set of reactor-specific safety functions and define the SSCs available or potentially available to perform these safety functions. This section describes the basis for defining the safety functions modeled in the MHTGR PRA and selecting the SSCs to be modeled in the performance of these safety functions.

As noted previously, the scope of SSCs to be included in the PRA includes all SSCs that perform either a required or supportive safety function for the radionuclide sources and barriers in the scope of the PRA. Since the PRA is performed initially, prior to the safety classification of SSCs, it is not initially known which modeled SSCs will perform a required safety function and of those which will be relied on in the Chapter 15 safety analyses.

The exhaustive set of IEs determined in Step 6 (Figure 3-6) is grouped according to the nature of the challenges to PRA safety functions. PRA safety functions have been defined in the context of a top-down logical structure starting with the high-level function of controlling the transport of radionuclides. Such transport is fundamentally controlled in the safety design approach by preserving the integrity of the radionuclide transport barriers identified in Table 3-2.

3.8.3 MHTGR Safety Functions and Supporting SSCs

Both inherent and engineered (other than inherent) safety features and SSCs are included in the design to perform the safety functions. The inherent features are the characteristics that are the direct consequence of the selection of materials and design features of the reactor fuel and core, moderator (for thermal reactor designs), and coolant. Engineered safety features are those introduced specifically to perform a safety function, and they may include both passive and active SSCs.

Consistent with good PRA practice, the safety functions modeled in the PRA include those required to meet the required safety functions, as well as SSCs included to meet availability and investment protection needs and serve DID roles by preventing and mitigating challenges to barriers and SSCs performing the required safety functions. The MHTGR safety design philosophy uses inherent safety features and passive SSCs to perform the required safety functions. Active SSCs are also provided for supportive safety functions as well as to meet plant investment protection and availability performance requirements. SSCs that serve both required and supportive safety functions are included in the PRA in order to capture a sufficiently complete set of safety function challenges and associated event sequences and to apply the principle of realistic PRA success criteria.

The process of using safety functions to develop the event sequences is fundamentally the same process as used in LWR PRAs. The need to model both safety and non-safety classified SSCs is also no different; only the functions and SSCs differ. Once the differences in safety functions and the SSCs that provide these functions are understood, the capability to review the PRA event sequence model is achieved.

The safety functions for the MHTGR, which are representative of all known modular HTGR designs include:

- Maintain control of radionuclides
    - Maintain control within fuel barriers
    - Maintain control within HPB
    - Maintain control within reactor building
- Control heat generation (reactivity)
- Control heat removal
- Control chemical attack
- Maintain core and reactor vessel geometry

A summary of the inherent features and passive SSCs along with the active SSCs that support or provide DID for the safety functions for an HTGR is provided in Table 3-3. The table shows design features representative of those under consideration for the NGNP. This indicates the types and scope of SSCs that would be modeled in the NGNP PRA. Each HTGR module includes one primary loop with a steam generator and a conventional steam turbine generator. It has a steam generator isolation and dump system to minimize water ingress to the primary following a steam generator tube failure and resides in a vented leak controlled reactor building with a recirculation filter heating, ventilation, and air-conditioning (HVAC) system.

Table 3-3. Major SSCs Modeled in the Example NGNP HTGR PRA

Safety Function	Inherent Features and Passive SSCs	Active SSCs*
Control Radio-nuclides	• Fuel barrier ▪ Fuel particle kernel ▪ Silicon carbide and pyrocarbon coatings of fuel particle ▪ Fuel matrix and fuel element graphite • HPB barrier • Reactor building barrier ▪ Retention capabilities of reactor building ▪ Reactor building pressure relief vents	• Primary system safety relief valves • Reactor building dampers (reclosure) • Reactor building HVAC filtration system • Steam generator isolation and dump system isolation valves
Control Heat Generation	• Strong negative temperature coefficient of reactivity • Gravity fall of control rods and reserve shutdown system absorber material	• Control and protection systems ▪ Operational control systems ▪ Investment protection system ▪ Reactor Protection System • Reactivity control systems ▪ Trip release of control rod drives ▪ Reserve shutdown system release of absorber material
Control Heat Removal	• Large thermal heat capacity • Passive core heat removal • Core size, power density, geometry • Core, uninsulated reactor vessel, and reactor cavity configuration • Passive Reactor Cavity Cooling System (RCCS) • Reactor building pressure relief vents	• Main Loop cooling systems via: ▪ Electric power conversion system ▪ Process steam system • Shutdown cooling system
Control Chemical Attack	• HPB high reliability piping and pressure vessels • HPB design minimize penetrations in top of reactor vessel • High purity specifications for inert helium coolant • Primary system safety valves • Reactor building pressure relief vents	• Reactor building vent dampers limit air ingress • Isolation valves in primary interfacing systems • HPS maintains high purity levels of helium coolant • Steam generator isolation and dump system
Maintain Core and Reactor Vessel Geometry	• Reactor core and structures • Reactor pressure vessel and structures • Passive RCCS maintains integrity of structures • Reactor building structure	• None

*Not shown in this table are support systems such as electric power systems, instrument and service air systems, and some of the man-machine interface systems.

Functional IE categories are defined by the nature of the challenge to safety functions. These categories are used to decide which different event sequence models need to be developed. The following list presents representative examples (not considered exhaustive) of functional IE categories being considered for the advanced non-LWR PRA for the sources of radioactive material inside the reactor vessel and the primary system pressure boundary:

- Plant transients with intact primary system HPB:
    - Main Loop and Shutdown Cooling System (SCS), still capable of forced cooling operation
    - Main Loop system failed, SCS still capable of operation
    - SCS failed, Main Loop system still capable of operation
    - Main Loop and SCS not capable of operation
- Energy conversion system transients with intact HPB and reactivity addition:
    - Control rod or group withdrawal
    - Overcooling transients
- Primary system HPB leaks and breaks:
    - HPB failures resulting in slow depressurization
    - HPB failures resulting in rapid depressurization
- HPB heat exchanger failures:
    - Steam generator tube leak
    - Steam generator tube rupture
    - SCS heat exchanger failure

Each of the above categories represents a unique challenge to the MHTGR required and supportive safety functions. These categories are used as a starting point for the development of event sequence models as described below.

Specific IEs or causes of IEs for each of the above categories can be defined having the same functional challenge to the safety functions. For example, one cause of a transient with the Main Loop system failed and the SCS still capable of operation (if the onsite diesel generator successfully starts) is a loss of offsite power. An example of a transient with the Main Loop and SCS still capable of operation is a Power Turbine Generator trip. Seismic events that do not cause a breach of the HPB are classified as power conversion system transients, while those that do are included in the HPB leaks and breaks category. To meet the requirements in the Advanced non-LWR PRA Standard, the comprehensive treatment of IEs and how they are dispositioned by screening and grouping will be documented according to applicable PRA guides and standards.

3.8.4 Development of Event Sequence Models

Once functional categories of the IEs are established, event sequence diagrams and event trees are developed to define event sequences resulting from each IE and initial condition to be modeled. The event trees will be quantified for each specific IE in each functional IE category in order to account for significant dependencies between the causes of the IE and the modeled SSC failure probabilities. The event tree top events will be derived in consideration of the SSCs provided to support each of the safety functions. The event sequences define the possible successes and failures of each SSC to implement each safety function to a sufficient extent to determine the event sequence end-states.

The treatment of operator actions in the modeling and quantification of event sequences follows the same process as for LWR PRAs. The following are the major differences in human reliability analysis (HRA) treatment in the modular HTGR PRA:

- Because of the safety design approach of the modular HTGR, there are few operator actions that must be fulfilled to achieve a safe, stable end-state to an event sequence.
- In general, the time windows available to implement the operator actions in the PRA model are very long. The application of existing HRA techniques that recognize the dependence of the human error rate on the time window may result in human error rates that are too small to be verifiable or appear credible. This is expected to result in a conservative treatment of human error rates in relation to that which would be considered realistic. It should not be viewed as a problem for the HTGR PRA because the PRA results are not that sensitive to the assumed human error rates and most of the important safety functions are fulfilled without need for time critical operator actions. Hence, the use of conservative human error rates is not expected to mask risk insights.
- Since there is less experience in performing HRA in PRAs for reactors such as the HTGR, it is expected that the uncertainties in the human error rates will be larger than found in typical LWR PRAs. For the same reasons cited above regarding the use of conservative human error rates, the assignment of large uncertainties should not be viewed to adversely impact the PRA results or their use in selecting LBEs.
- Because the PRA provides input in the selection of LBEs and the greater reliance on inherent and passive means to fulfill safety functions in the HTGR, there will be increased emphasis on the treatment of human errors of commission in the HTGR PRA.
- At the early design stage versions of the PRA, many of the details of the emergency operating procedures, man-machine interface, and human factors engineering model will be unknown. This will be taken into account in the human error rate uncertainty analysis and will tend to increase the uncertainties. As noted above, this is not expected to cause a problem in terms of masking risk insights or adversely impacting the capability of the PRA to support LBE selection.

Figure 3-7 depicts the event sequence modeling framework for the advanced non-LWR. This framework includes the following elements:

- IE in the context of a plant operating state
- Plant response to IE
- Response of the reactor building and associated SSCs
- Factors influencing the end-state, including achievement of success criteria and mechanistic source terms

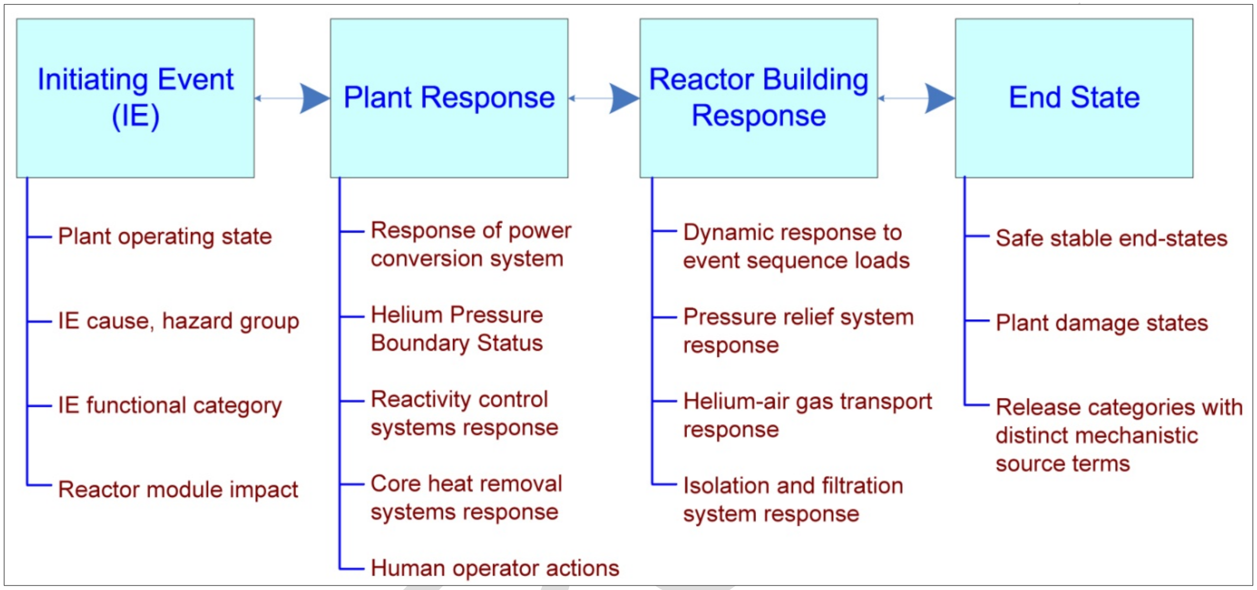

Initiating Event (IE)
Plant operating state
IE cause, hazard group
IE functional category
Reactor module impact
Plant Response
Response of power conversion system
Helium Pressure Boundary Status
Reactivity control systems response
Core heat removal systems response
Human operator actions
Reactor Building Response
Dynamic response to event sequence loads
Pressure relief system response
Helium-air gas transport response
Isolation and filtration system response
End State
Safe stable end-states
Plant damage states
Release categories with distinct mechanistic source terms

Figure 3-7. Event Sequence Modeling Framework for a Modular HTGR PRA

The causes of the IEs depicted in Figure 3-7 include internal plant hardware failures, human errors, internal plant hazards such as fires and floods, and external hazards such as seismic events and transportation accidents. The responses of the plant and reactor building functions include the responses of SSCs and the human operators that are involved in the performance of or failure to perform each function. Human responses include favorable or unfavorable acts and errors of omission and commission.

Using the high-level logic of Figure 3-7, the framework for developing the event sequences for an HTGR includes the following elements:

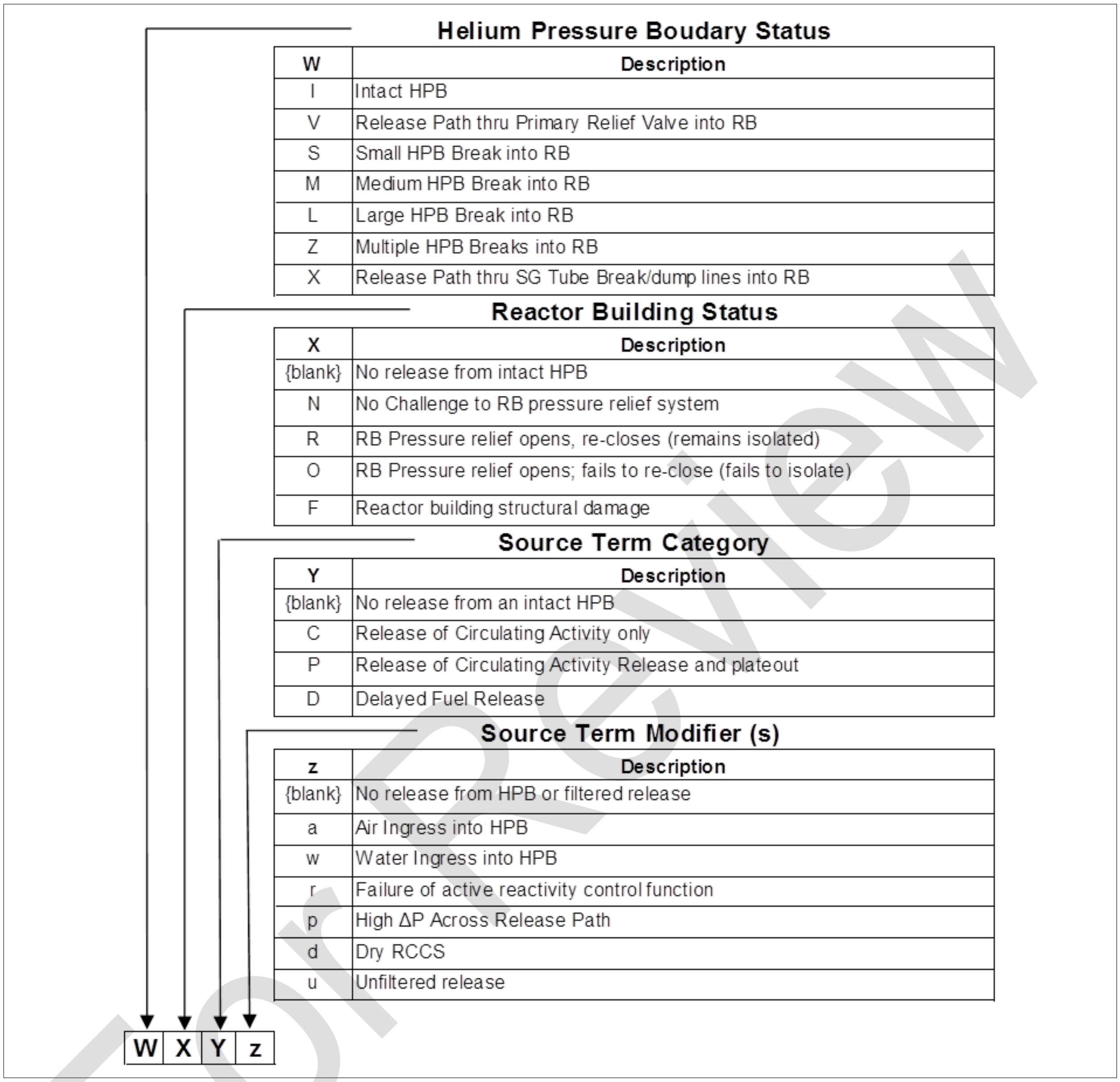
- Definition of initial plant conditions
- IE
- Cause of the IE
- IE functional category
- Reactor module impact (single versus multiple modules)
- Response of the systems for reactivity control

- Response of primary circulators to IE
- Status of the Helium Pressure Boundary including any mitigating actions
- Response of the core heat removal systems
- Core oxidation status
- Status of fuel in each affected module
- Response of the reactor building(s)
- Early response to IE
- Long term response including mitigating actions
- Source term characteristics for each affected module

3.8.5 Event Sequence End States for Modular HTGR PRA

The thought process used to apply this framework in the development of the event sequence diagrams (ESDs) and event trees for each IE is to ask sufficient questions to resolve the end states of each sequence. These end states start with the states that define the safe termination of the event with no damage or release beyond that which is consequential to the IE and include all the possible plant damage states and states of radioactive release. In previous HTGR PRAs, an end state coding scheme was developed to capture all the factors that characterize a unique modular HTGR end state.

The coding scheme in Figure 3-8 was developed for previous modular HTGR PRAs including the PBMR PRA in South Africa and serves as a starting point for future HTGR PRA end states. Each end state code is defined by three capital letters and one or more small letters. The first letter defines the status of the primary HPB at the end of the accident sequence, the second defines the response of the reactor building (RB) and associated SSCs, and the third letter defines the components of the radioactive release source term from the plant site, if the sequence has a release. Small letters at the end are source term modifiers. These source term end states codes facilitate grouping of event sequences having the same or similar consequences which is important because the event trees will eventually capture hundreds to thousands of event sequences.



Helium Pressure Boudary Status

W	Description
I	Intact HPB
V	Release Path thru Primary Relief Valve into RB
S	Small HPB Break into RB
M	Medium HPB Break into RB
L	Large HPB Break into RB
Z	Multiple HPB Breaks into RB
X	Release Path thru SG Tube Break/dump lines into RB

Reactor Building Status

X	Description
{blank}	No release from intact HPB
N	No Challenge to RB pressure relief system
R	RB Pressure relief opens, re-closes (remains isolated)
O	RB Pressure relief opens; fails to re-close (fails to isolate)
F	Reactor building structural damage

Source Term Category

Y	Description
{blank}	No release from an intact HPB
C	Release of Circulating Activity only
P	Release of Circulating Activity Release and plateout
D	Delayed Fuel Release

Source Term Modifier (s)

z	Description
{blank}	No release from HPB or filtered release
a	Air Ingress into HPB
w	Water Ingress into HPB
r	Failure of active reactivity control function
p	High ΔP Across Release Path
d	Dry RCCS
u	Unfiltered release

Figure 3-8. End State Codes for HTGR Event Sequences

3.8.6 LBEs as Event Sequence Families

In selecting LBEs, event sequence families are used to group together two or more event sequences when the sequences have a common IE, 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 IE, safety function responses, and end-state. The end-state for a multi-module plant includes the number of reactor modules involved in any releases for the event sequence.

- 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. Event sequences with similar but not identical safety function responses are not combined when such combination would mask the definition of unique challenges to the SSCs that perform safety functions.
- 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. In this case, the individual event sequences are associated with a specific reactor module and the family groups them together for the entire multi-module plant.
- Each event tree IE 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. Hence each event sequence is already a family of event sequences when the information in the fault trees is taken into account.
- The frequency of the LBE defined by the accident family is the linear sum of the individual event sequence frequencies. The frequency units are events per plant-year. This provides a common frequency basis to compare and combine different types of sequences involving different numbers of reactor modules, and different plant operating states.

Without the use of event sequence families, the level of detail in the definition of the IE categories and decisions to balance the level of detail between the event trees and fault trees may inadvertently impact the classification of an individual event sequence as an AOO, DBE, or BDBE. By aggregating the sequences into the event sequence families, the decisions made in structuring the event sequence model do not impact the LBE classification. A discussion of how event sequence families are used to define LBEs is provided in the LBE selection white paper.[2]

3.8.7 Example Modular HTGR Event Sequence Model for Slow Depressurization Event

The purpose of this section is to present an example of the event sequence development for one IE that is common to all modular HTGR designs. The event sequence development example includes a definition of the IE, the development of an event sequence diagram, and an event tree which quantifies the event sequence frequencies, assignment of end states, and a classification into LBE categories. This is an example of a PRA model that was developed in an early stage of the conceptual design of a modular HTGR and is provided to show the level of detail of an early stage PRA that is sufficient to develop an initial set of LBEs.

3.8.7.1 Definition of Initiating Event

The IE for this example is defined as a small depressurization event on one and only one HTGR module. This event is a leak or breach in the primary system HPB with an equivalent break size up to 10 mm diameter. Such an event, if not isolated, would result in a slow depressurization of the primary system so that if forced circulation cooling is lost there would still be a positive pressure in the primary system when peak core temperatures are reached during a depressurized condition cooldown. This event has a relatively high frequency of occurrence. Possible causes include leaks or breaks in HPB piping or welds connecting pipes and vessels in the primary system pressure boundary. It would be expected that the module could be restarted and return to power once the breach is corrected, however a relatively long outage may be needed to repair the affected component.

3.8.7.2 Safety Design Mitigation Strategy

It is assumed in the development of the event sequences for this event that the HTGR will be designed to implement the following event mitigation strategy. This strategy is developed through collaboration between the design team and the PRA team:

- Some parts of the active HPB, such as Helium Purification System piping, have isolation valves that can be closed to terminate the leak, depending on the location of the leak relative to the isolation valves. It is assumed in the initial PRA development that there will be means to detect the leak and isolate it via automatic or manual action. Risk insights from the initial PRA development for this event will assist the design team to incorporate capabilities in the design to detect and isolate leaks.
- For very slow leaks and rates of depressurization, there should be a means of pumping down the system to reduce the system pressure in the event there is a loss of forced cooling. Risk insights from this initial PRA development will help the designers evaluate and decide on the helium pump down capabilities.
- The reactor will be shutdown to reduce heat generation either by operator action or reactor trip via Investment Protection System (IPS) or Reactor Protection System (RPS).
- Forced cooling of the affected reactor can be maintained by the Main Heat Transport System or Startup/Shutdown (SU/SD) System until the system is fully cooled down and depressurized.
- If forced cooling cannot be established, passive cooling via the RCCS will be provided until forced cooling and module restart is affected.
- An RB HVAC system can be used to reduce the environmental release source term. While such a system may not be necessary, it is included in the sequence development to support future RB design options studies.

3.8.7.3 Key Assumptions Regarding Plant Response and SSC Capabilities

The following assumptions are made to support event sequence development during the conceptual design and will be replaced by appropriate analyses as the design matures.

- The plant control systems will be designed so that expected plant transients such as slow depressurization on one module do not disturb or adversely affect operation of the remaining modules.

- The rate of depressurization for this category of small leaks is insufficient to challenge the reactor building pressure relief system; however, an RB HVAC system can be used to filter any releases to the environment.
- Although the response of the reactivity control systems is identified in the sequence development, due to the negative temperature coefficient, the reactor power will always match the core heat removal; hence, failure to trip the reactor has little if any impact on the resulting end state.

3.8.7.4 Event Sequence Diagram Development

The first step in the development of the event sequence model for slow or small depressurization is to develop an ESD, which is often used in a PRA as a precursor to an event tree development. The ESD symbols used in this example are shown in Figure 3-9, and the acronyms used in the ESD and event tree are defined in Table 3-4.

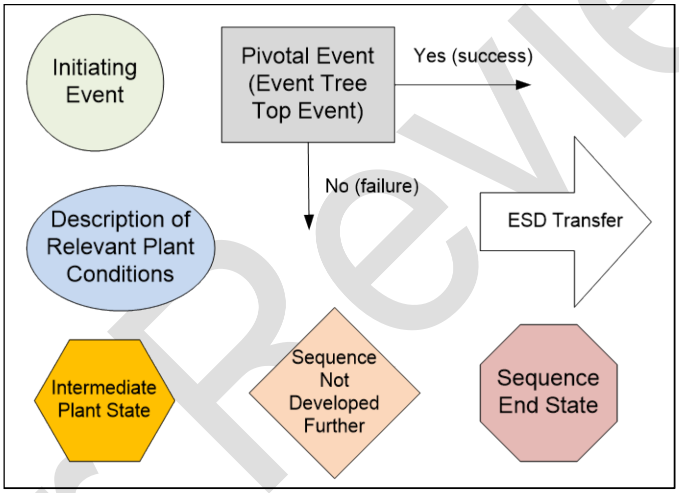

Initiating Event
Pivotal Event (Event Tree Top Event)
Yes (success)
No (failure)
Description of Relevant Plant Conditions
ESD Transfer
Intermediate Plant State
Sequence Not Developed Further
Sequence End State

Figure 3-9. Event Sequence Diagram Symbols

Table 3-4. Acronyms Used in ESD

CA—Circulating primary coolant activity
CC—Conduction cool down (RCCS works)
CG—Conduction to ground (RCCS fails)
DCC—Depressurized conduction cool down (RCCS works)
DCG—Depressurized conduction to ground (RCCS fails)
D(LO)FC—Depressurized (Loss of) forced circulation cooling
FC—Forced circulation cooling
HPB—PHTS Helium Pressure Boundary
MLTP—heat transport path via ML
ML—Main Loop Heat Transport System
IPS—Investment Protection System
OCS—Operational Plant Control System
PCC—Pressurized conduction cooling (RCCS works)
PCG—Pressurized conduction to ground (RCCS fails)
PFC—Pressurized forced cooling
RCCS—Reactor Cavity Cooling System
RCTP—Heat transport path via RCCS
RPS—Reactor Protection System
RSS—Reserve Shutdown System
RTF—Reactor Trip Failure Transient
SCP—Heat transport path via SU/SD System
SDG—Standby Diesel Generator
SCS—Shutdown Cooling System
SG—Steam Generator
SV—Primary Heat Transport System Safety Valve

Figure 3-10 and Figure 3-11 provide the ESD for the small or slow depressurization event for the HTGR. If the leak is successfully isolated in Event 2, the Operational Plant Control System (OCS) is designed to maintain power operation with a minor increase in normal leakage, and the sequence successfully terminates. With failure to isolate in Event 2, there is a slow depressurization of the reactor and signals to trip the reactor from the IPS and RPS at different setpoints for high radiation in the reactor building or low system pressure. The automatic trip functions modeled in Events 4a and 4b are backed up by manual actions to trip the reactor in Event 4c. If the control rods are not inserted, the reactor power will follow the core heat removal rate. In Events 5 and 6, consideration is given as to whether forced circulation cooling is maintained using either the Main Loops (MLs) or SCS. For modular HTGRs, when forced cooling is maintained the only source term available for release to the reactor building is the circulating activity that exists in normal operation. In the event that forced cooling is not maintained, there will be small delayed fuel release source term which will occur when peak core temperatures are reached more than 24 hours after the IE. The release of this delayed fuel release to the reactor building can be mitigated by pumping down the HPB which reduces the pressure driving force to expel radionuclides out the break into the RB. On Page 2 (Figure 3-11) of the ESD, the responses of the RCCS and the reactor building HVAC systems are considered which response will modify the magnitude of the source term from the reactor building.

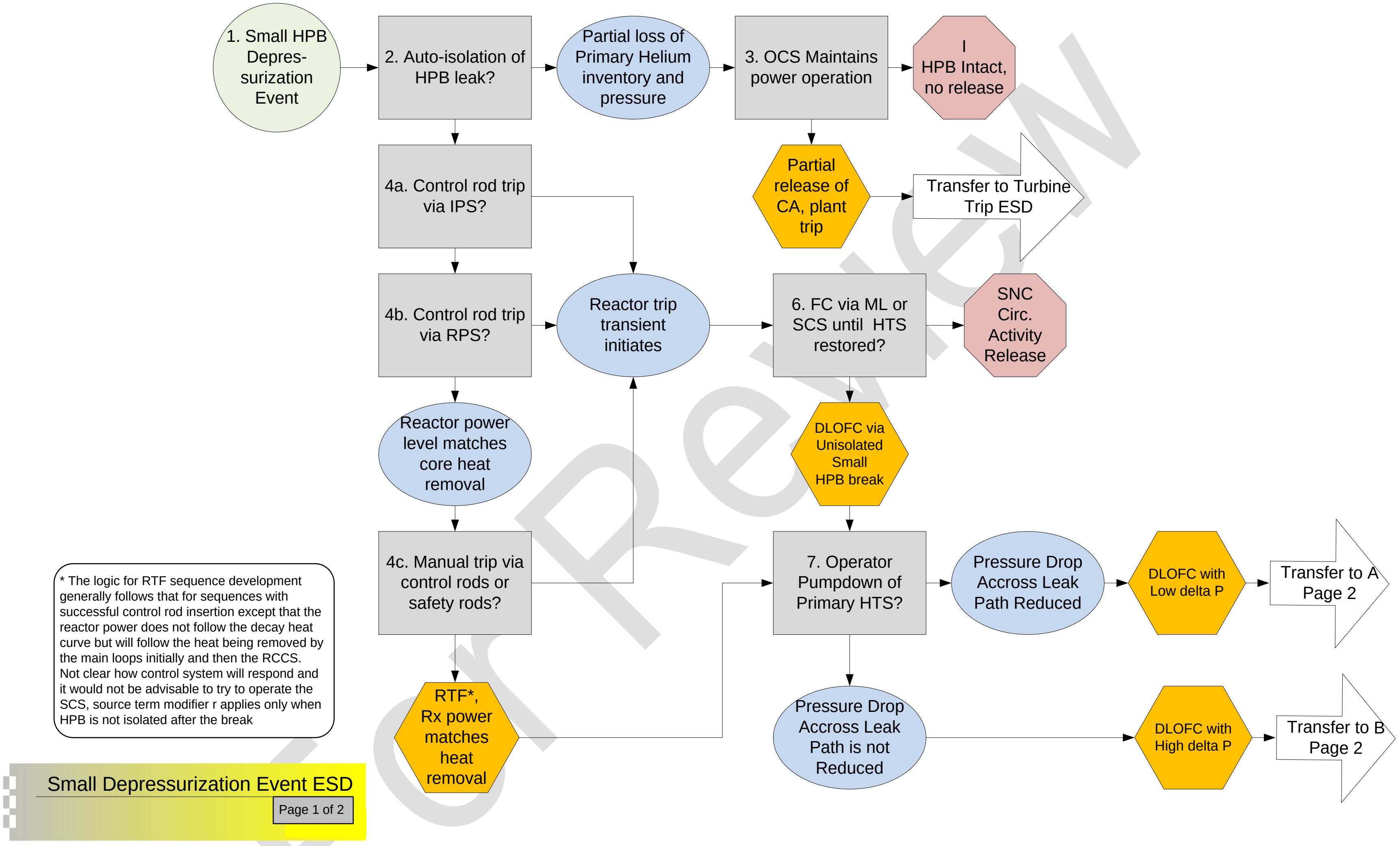

1. Small HPB Depres-surization Event
2. Auto-isolation of HPB leak?
Partial loss of Primary Helium inventory and pressure
3. OCS Maintains power operation
I HPB Intact, no release
4a. Control rod trip via IPS?
Partial release of CA, plant trip
Transfer to Turbine Trip ESD
4b. Control rod trip via RPS?
Reactor trip transient initiates
6. FC via ML or SCS until HTS restored?
SNC Circ. Activity Release
Reactor power level matches core heat removal
DLOFC via Unisolated Small HPB break
4c. Manual trip via control rods or safety rods?
7. Operator Pumpdown of Primary HTS?
Pressure Drop Accross Leak Path Reduced
DLOFC with Low delta P
Transfer to A Page 2
* The logic for RTF sequence development generally follows that for sequences with successful control rod insertion except that the reactor power does not follow the decay heat curve but will follow the heat being removed by the main loops initially and then the RCCS. Not clear how control system will respond and it would not be advisable to try to operate the SCS, source term modifier r applies only when HPB is not isolated after the break
RTF*, Rx power matches heat removal
Pressure Drop Accross Leak Path is not Reduced
DLOFC with High delta P
Transfer to B Page 2
Small Depressurization Event ESD
Page 1 of 2

Figure 3-10. Event Sequence Diagram for HTGR Small (Slow) Depressurization Event (Page 1 of 2)

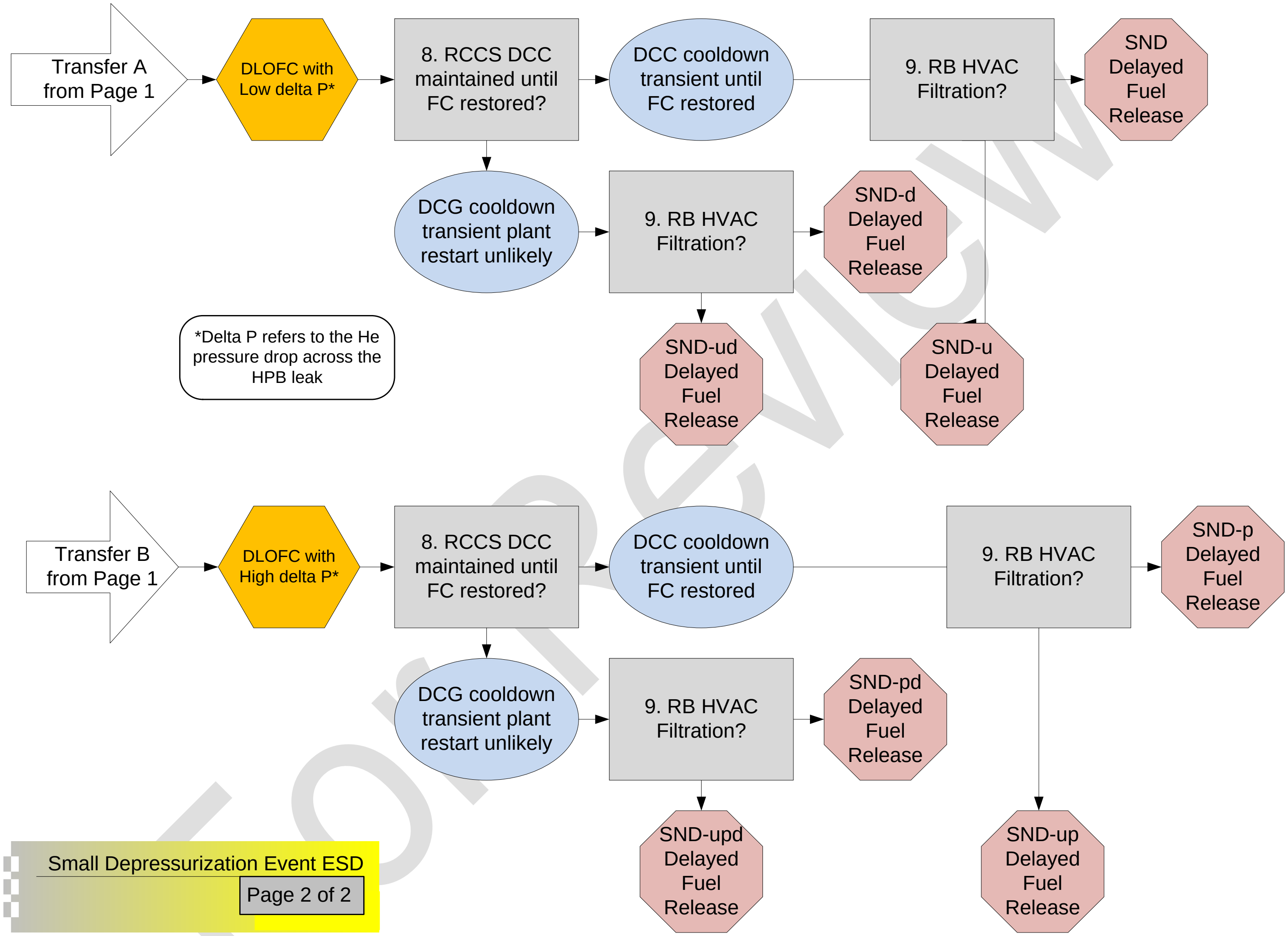
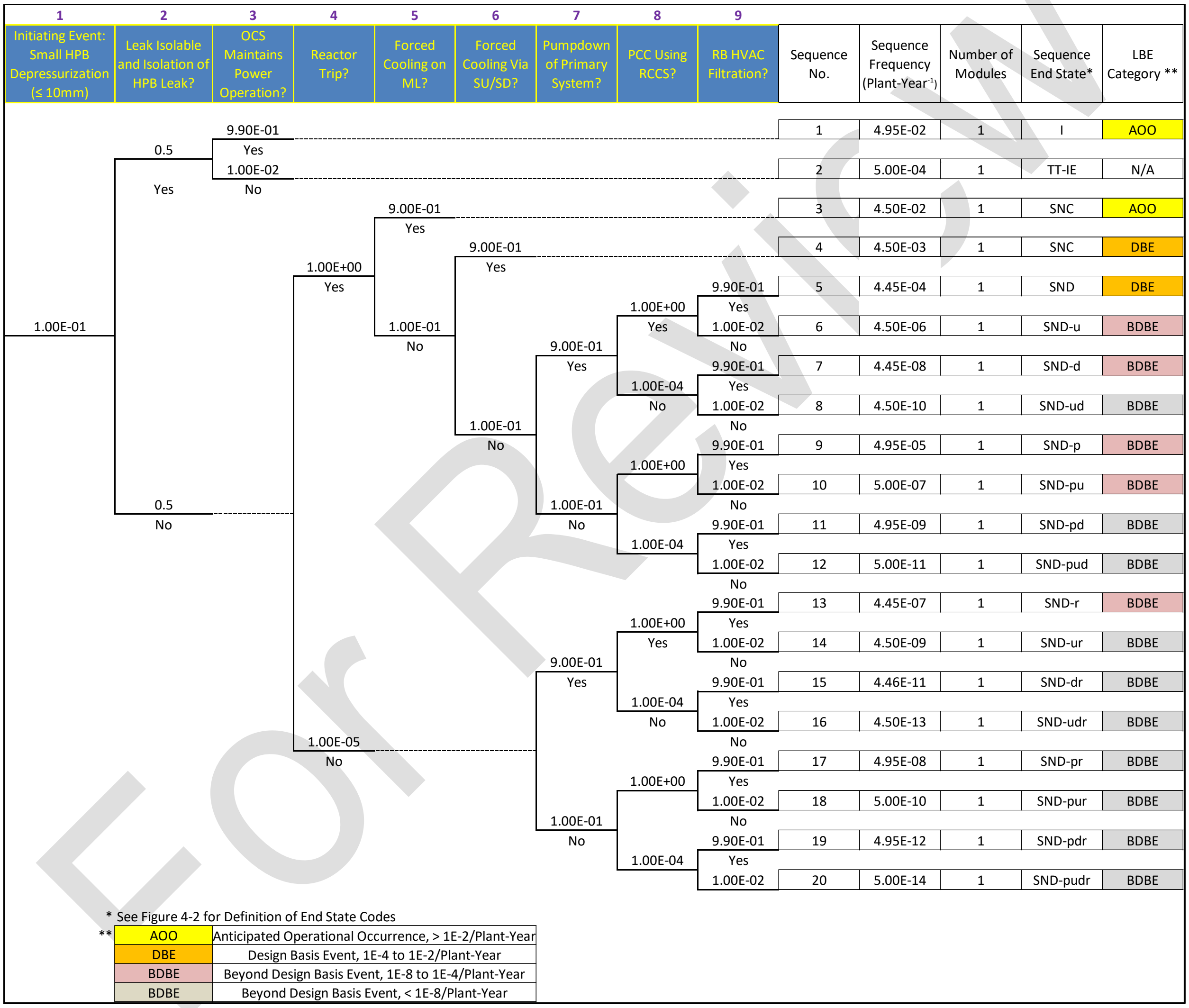

Transfer A from Page 1
DLOFC with Low delta P*
8. RCCS DCC maintained until FC restored?
DCC cooldown transient until FC restored
9. RB HVAC Filtration?
SND Delayed Fuel Release
DCG cooldown transient plant restart unlikely
9. RB HVAC Filtration?
SND-d Delayed Fuel Release
SND-ud Delayed Fuel Release
SND-u Delayed Fuel Release
*Delta P refers to the He pressure drop across the HPB leak
Transfer B from Page 1
DLOFC with High delta P*
8. RCCS DCC maintained until FC restored?
DCC cooldown transient until FC restored
9. RB HVAC Filtration?
SND-p Delayed Fuel Release
DCG cooldown transient plant restart unlikely
9. RB HVAC Filtration?
SND-pd Delayed Fuel Release
SND-upd Delayed Fuel Release
SND-up Delayed Fuel Release
Small Depressurization Event ESD
Page 2 of 2

Figure 3-11. Event Sequence Diagram for HTGR Small (Slow) Depressurization Event (Page 2 of 2)

The event tree for this small depressurization IE is shown in Figure 3-12. This event tree example was developed at an early stage in the conceptual design. The quantification of the event tree sequences was made using engineering judgments that were sufficient to obtain order of magnitude estimates. As the design matures, these estimates will be replaced with appropriately detailed system fault tree models and human reliability analyses, and data analyses sufficient to meet PRA standard requirements. However, these estimates provide insights to the design in the form of a preliminary list of LBEs. Table 3-5 lists the preliminary LBEs developed using this example.



Sequence No.	Sequence Frequency (Plant-Year^{-1})	Number of Modules	Sequence End State*	LBE Category **
1	4.95E-02	1	I	AOO
2	5.00E-04	1	TT-IE	N/A
3	4.50E-02	1	SNC	AOO
4	4.50E-03	1	SNC	DBE
5	4.45E-04	1	SND	DBE
6	4.50E-06	1	SND-u	BDBE
7	4.45E-08	1	SND-d	BDBE
8	4.50E-10	1	SND-ud	BDBE
9	4.95E-05	1	SND-p	BDBE
10	5.00E-07	1	SND-pu	BDBE
11	4.95E-09	1	SND-pd	BDBE
12	5.00E-11	1	SND-pud	BDBE
13	4.45E-07	1	SND-r	BDBE
14	4.50E-09	1	SND-ur	BDBE
15	4.46E-11	1	SND-dr	BDBE
16	4.50E-13	1	SND-udr	BDBE
17	4.95E-08	1	SND-pr	BDBE
18	5.00E-10	1	SND-pur	BDBE
19	4.95E-12	1	SND-pdr	BDBE
20	5.00E-14	1	SND-pudr	BDBE

* See Figure 4-2 for Definition of End State Codes

**		
	AOO	Anticipated Operational Occurrence, > 1E-2/Plant-Year
	DBE	Design Basis Event, 1E-4 to 1E-2/Plant-Year
	BDBE	Beyond Design Basis Event, 1E-8 to 1E-4/Plant-Year
	BDBE	Beyond Design Basis Event, < 1E-8/Plant-Year

Figure 3-12. Event Tree for HTGR Small (Slow) Depressurization Event

An example of a more detailed PRA based on the conceptual design of the MHTGR is found in Reference [4]. The results of that PRA are used to illustrate the LMP approach of selecting and evaluating LBEs in Reference [2].

Table 3-5. LBEs for Small Depressurization Initiating Event in a Modular HTGR

LBE No.	LBE Type	Plant Response	Frequency, Plant-Year	End State
SD-1	AOO	Leak isolated, OCS maintains power operation	4.95E-02	I
SD-3	AOO	Fail to isolate leak, reactor trip, forced cooldown on ML	4.50E-02	SNC
SD-4	DBE	Fail to isolate leak, reactor trip, ML failure, forced cooldown on SCS	4.50E-03	SNC
SD-5	DBE	Fail to isolate leak, reactor trip, ML failure, SU/SD System failure, primary pumpdown, conduction cooldown via RCCS, RB filtration	4.45E-04	SND
SD-6	BDBE	Fail to isolate leak, reactor trip, ML failure, SU/SD System failure, primary pumpdown, conduction cooldown via RCCS, RB filtration failure	4.50E-06	SND-u
SD-9	BDBE	Fail to isolate leak, reactor trip, ML failure, SU/SD System failure, primary pumpdown failure, conduction cooldown via RCCS, RB filtration	4.95E-05	SND-p
SD-10	BDBE	Fail to isolate leak, reactor trip, ML failure, SU/SD System failure, primary pumpdown failure, conduction cooldown via RCCS, RB filtration failure	5.00E-07	SND-p
SD-13	BDBE	Fail to isolate leak, reactor trip failure, primary pumpdown, conduction cooldown via RCCS, RB filtration	4.45E-07	SND-r

3.8.8 Example HTGR PRA Results

Example HTGR results evaluated for the PBMR in South Africa are shown in Figure 3-13 in comparison with the frequency-dose criteria that were defined for the NGNP project.[1] The results reflect quantified uncertainties in both the frequencies and site boundary doses (using the unit of Total Effective Dose Equivalent, or TEDE) in comparison with the Top Level Regulatory criteria for acceptable frequencies and consequences that were proposed in the NGNP LBE selection white paper.[6] The site boundary dose consequence uncertainties are based on mechanistic source terms that were evaluated for PBMR event sequences grouped into LBE categories which are called for in the ASME/ANS PRA Standard.[42]

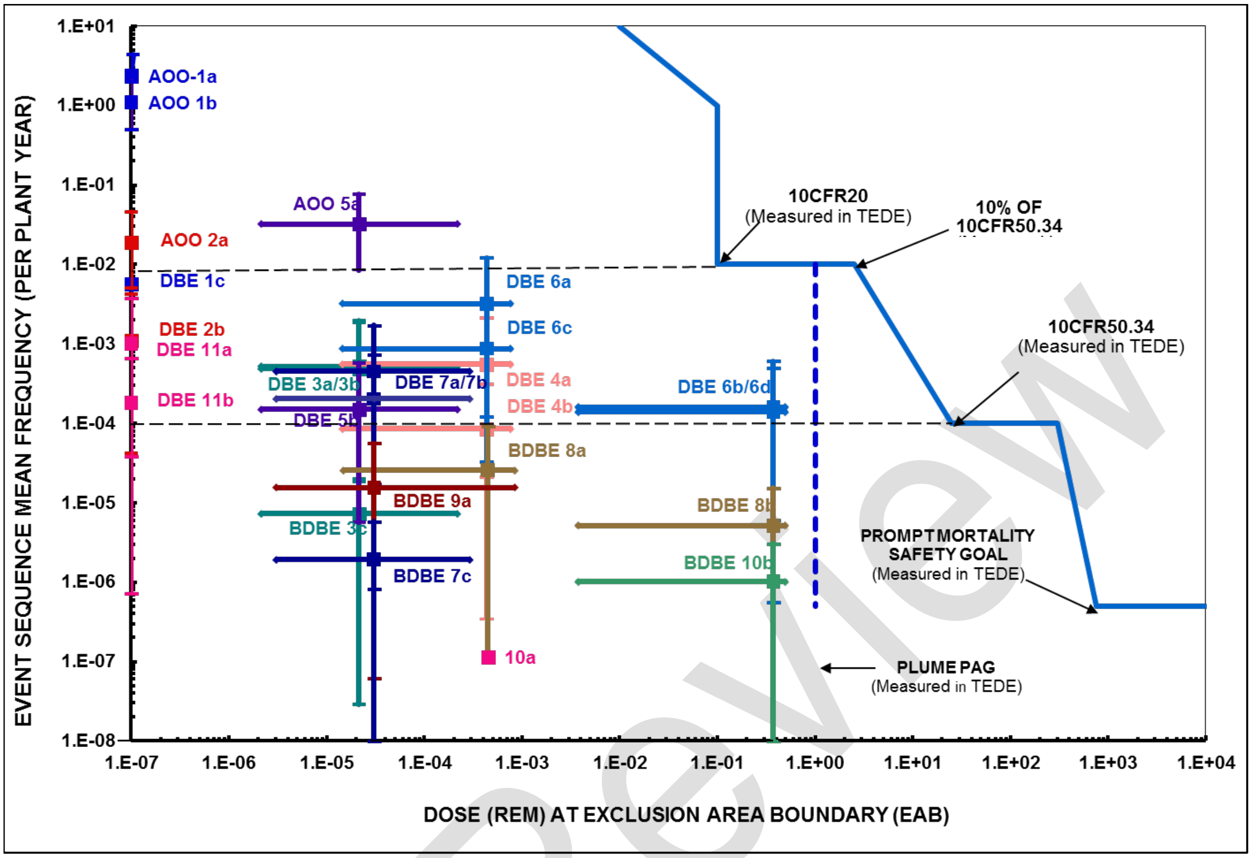


Figure 3-13. PBMR LBE Results Compared with NGNP Frequency-Dose Criteria

In the LMP LBE white paper on the selection and evaluation of LBEs,[2] PRA results from Reference [4] are used to illustrate how LBEs are developed from a completed conceptual design of the MHTGR. The reader is referred to that PRA for details on how the event sequences are defined and quantified, how the plant response and mechanistic source terms were evaluated, and radiological consequences determined.

3.9 Example PRA Development for PRISM Sodium-Cooled Fast Reactor

The purpose of this section is to summarize how the technology-inclusive approach to developing a PRA model for an advanced non-LWR is implemented for an SFR design. The example is based on results of a PRA performed on the PRISM sodium-cooled fast reactor, whose design is documented in Reference [40]. This example develops PRA model elements for a plant comprised of two reactor modules and includes both single module and multi-module accidents. Accident sequences involving non-core sources of radioactive material are not included in this example.

Please note that, unlike the MHTGR example presented in the previous section, which was developed to support the identification of LBEs and safety classification of SSCs, the PRISM PRA was developed to pilot the ASME/ANS PRA Standard for Advanced non-LWRs,[43] and was focused on the risk assessment of severe accidents. However, the applicability of the TI PRA approach for the LMP is aptly demonstrated.

The key characteristics of the SFR design include:

- The PRISM core resides in a pool reactor with sodium coolant and a cover gas space at the top of the vessel at essentially atmospheric pressure; therefore, high energy releases of primary coolant from the vessel are not physically possible. Because there are no reactor vessel penetrations below the top of the sodium level, line breaks that would lead to loss of coolant accidents (LOCAs) are not possible either.
- The intermediate heat transfer loop separates the primary sodium core coolant from the secondary sodium coolant whose heat is transferred to the Steam Generator.
- The intermediate heat transfer loop is elevated above the primary loop to provide enough pressure head to prevent migration of radioactive or contaminated primary sodium to the intermediate loop or the environment following an unexpected leak of an intermediate loop heat exchanger tube.
- Metal fuel with a sodium bond inside the cladding improves heat transfer and allows for axial expansion to significantly reduce fuel-clad stress.
- Sodium is chemically compatible with the fuel, cladding, vessel and piping surfaces, thus eliminating corrosive decay mechanisms.
- Passive negative reactivity feedback mechanisms that reduce reactor power or limit its increase to safe levels even under anticipated transients with failure of the active scram system. These mechanisms respond to increases in the system temperature (Doppler broadening of the non-fission absorption neutron cross section, and thermal expansion of the fuel, control rods, and core support structure).
- Given a failure of the normal heat removal path through the condenser, passive shutdown heat removal is available by the continuously operating and monitored Reactor Vessel Auxiliary Cooling System (RVACS) that relies only on the natural circulation of the primary sodium coolant and atmospheric air to remove decay heat.

3.9.1 Systematic Search for SFR Initiating Events

The IE analysis is performed to identify perturbations that could occur during any plant operating state (POS), that challenge plant control and safety systems, whose failures could potentially lead to undesirable plant conditions, including radioactive material release. IEs include transients, losses of offsite power and special initiator groups. IE identification is based on review of industry PRAs, guidance documents, and design experience.

The IEs are initially limited to at-power internal events, that is, those IEs occurring during power operation either as a direct result of equipment failure, or as the result of errors while performing maintenance, testing, or operator action. Then, IEs occurring as a result of external hazards such as seismic events and those occurring during shutdown POSs are identified and included for consideration.

A systematic approach is used to identify events that challenge normal plant operation and require successful mitigation to prevent radionuclide release. This includes an evaluation of previously identified LWR and non-LWR IEs that are also applicable to PRISM, and an assessment of the failure modes and effects of systems that are unique to the PRISM design.

Individual IEs that require similar response from front line and auxiliary systems and operators are combined into IE groups. Combining IEs into groups reduces the number of event trees that need to be developed. In grouping IEs, the events must be similar in terms of plant response, success criteria, accident progression timing (including time available for mitigating systems and operator actions to be performed), and the effect of the event on the operability and performance of mitigating systems and plant operators. The following IE groups are thus identified:

- Transient overpower
- Loss of primary forced flow (LOF)
- Intermediate Heat Exchanger (IHX) bypass leak
- Intermediate Heat Transport System (IHTS) leak
- Steam Generator Tube Rupture
- Nuclear Steam Supply System (NSSS) and General Transient faults, which have the sub-groups:
    - NSSS transients
    - Turbine / balance-of-plant (BOP) transient faults
    - BOP / Loss of Heat Sink faults
    - Loss of Offsite Power
    - Core Faults
    - Reactor Vessel Leak

3.9.2 PRISM Safety Functions

The PRISM reactor design has five concentric barriers that separate core radionuclide material from the environment.

- The metal fuel retains many radionuclides (i.e. plutonium, neptunium) within its matrix as long as the fuel has not melted.
- The fuel cladding around the fuel provides a barrier for gaseous fission products (i.e. xenon, krypton) as long as the cladding is intact.

- The sodium coolant acts as a third radionuclide barrier by retaining fission products either by chemical solubility or adsorption mechanisms.
- The reactor vessel is the radionuclide barrier for fission gases that are released by the sodium to the cover gas space which is at or near atmospheric pressure.
- The containment is the final radionuclide barrier. A filtered reactor building ventilation system could also possibly mitigate radioactive release to the atmosphere, but it is not credited in this analysis. A full application of the LMP approach to PRA for input to selection of LBEs would require that all systems capable of preventing or mitigating a release be included.

A set of four key safety functions for the PRISM design prevent or mitigate damage to the radionuclide release barriers (i.e., reactivity control, core flow, heat removal, and radionuclide confinement). As IEs present challenges to one or more release barriers, various systems are relied upon to provide safety functions that protect against those barrier challenges. The plant's response systems are characterized by the functions listed below and illustrated in Figure 3-14.

- The reactivity control function influences the amount of heat being generated within the reactor core, which dictates the rate at which energy must be removed from the core. The success criterion of the reactivity control function is to reduce core power quickly enough to match core flow or heat removal faults in the short-term, preventing damage to the fuel matrix and fuel cladding barriers. Plant features that satisfy this function are the control rod drive system, inherent reactivity feedbacks (Doppler broadening, changes in sodium density and metallic fuel expansion and contraction), gas expansion modules, and the backup ultimate shutdown system control rods.
- The core flow function transports heat from the core to the primary sodium coolant. The success criterion of the core flow function is to provide enough cooling to match power, thus preventing damage to the fuel matrix and fuel cladding barriers due to overheating. The electromagnetic (EM) pump coastdown machines for unprotected loss of flow scenarios, and natural circulation for the remaining scenarios fulfill this safety function.
- The primary sodium heat removal function rejects the heat transported to the primary sodium coolant away from the reactor vessel. The success criterion of the primary sodium heat removal function is to remove enough heat to prevent damage in the long-term to the fuel cladding and reactor vessel barriers. The shutdown heat can be removed by three systems: (1) the main condenser, (2) the auxiliary (steam generator to air) cooling system, and (3) the passive RVACS. Supporting features/systems include the intermediate heat transfer loop through the Steam Generator, and tripping the breakers to the primary and intermediate EM pumps to eliminate heat from pumping power.
- The confinement function provides the integrity of the final radionuclide release barriers, namely the reactor vessel and the containment. Any failures of the fuel matrix or cladding that result in fission products released to the cover gas space are mitigated by the vessel head and the containment. The success criterion of the confinement function is a release of no more than the design basis leakage rate of radionuclides from the vessel head and containment. Metal-water detection, steam generator (SG) isolation and blowdown, vessel

head isolation, and containment isolation are key systems that work together to fulfill this function.

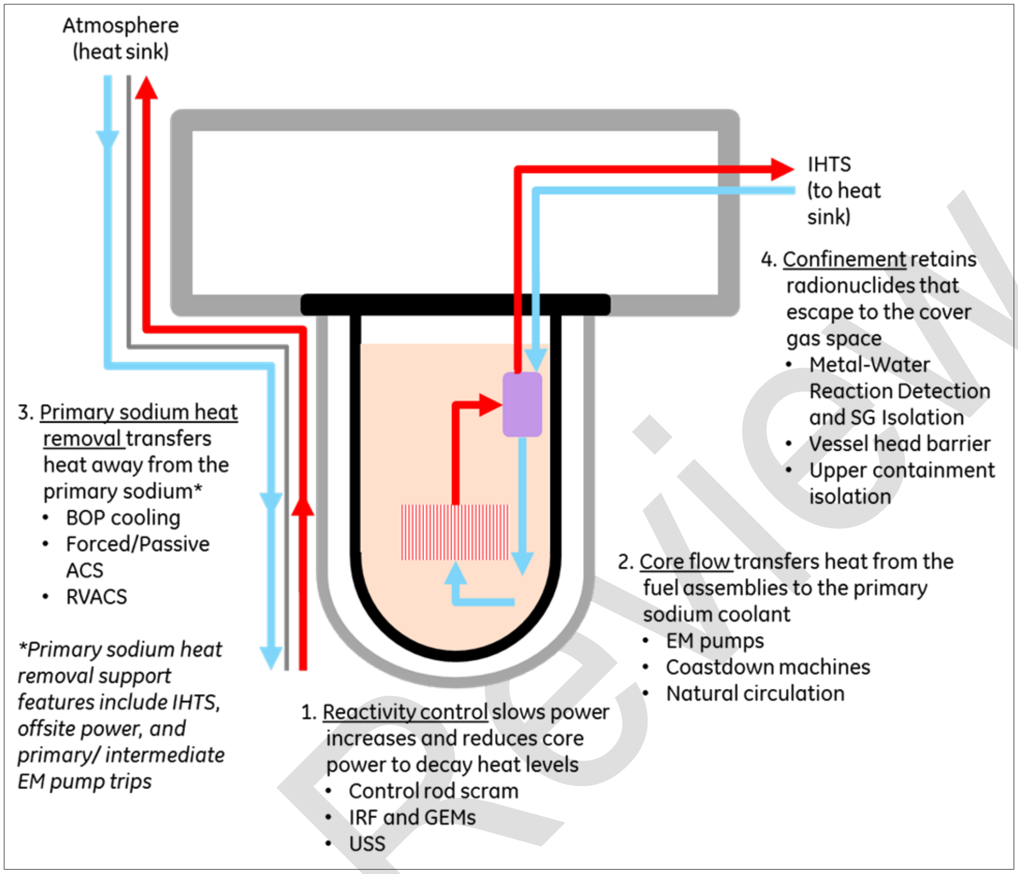

Atmosphere
(heat sink)
IHTS
(to heat
sink)
4. Confinement retains radionuclides that escape to the cover gas space
• Metal-Water Reaction Detection and SG Isolation
• Vessel head barrier
• Upper containment isolation
3. Primary sodium heat removal transfers heat away from the primary sodium*
• BOP cooling
• Forced/Passive ACS
• RVACS
*Primary sodium heat removal support features include IHTS, offsite power, and primary/ intermediate EM pump trips
2. Core flow transfers heat from the fuel assemblies to the primary sodium coolant
• EM pumps
• Coastdown machines
• Natural circulation
1. Reactivity control slows power increases and reduces core power to decay heat levels
• Control rod scram
• IRF and GEMs
• USS

Figure 3-14. PRISM Safety Functions

3.9.3 PRISM Safety Functions and Supporting System Development

Event trees developed for the PRISM PRA follow the standard practices, as described below. Event tree node success or failure is informed by the Success Criteria analysis. Success criteria are developed for active and passive systems performing safety functions modeled in the PRISM PRA. Unlike the success criteria for LWR PRAs, which are rooted in the known CDF risk surrogate, the PRISM PRA bases the success or failure of mitigating plant features on the potential for release of radionuclide material from the core with transport to the environment.

Active systems, such as Auxiliary Cooling Systems (ACSs), BOP cooling and electrical power are treated by traditional reliability analysis with failures and design basis performance characteristics identified for each analyzed system. In addition, passive systems that perform or support key safety functions are given special reliability treatment within the success criteria analysis because they often have no components that must move or change state to provide the safety function. They experience degrees of failure or degradation that range from no plant damage to higher, less probable levels of plant damage.

The success criteria for the PRISM design include the barriers and mitigating systems needed to prevent radiological releases based on the identification of possible release categories. Each release category consists of a quantified level of fuel barrier damage and a combination of intact and failed confinement release barriers. For each event sequence modeled, plant parameters are defined with various thresholds that represent the different release categories that are possible for that sequence. For those categories with a release, radiological consequences are evaluated.

The robust PRISM design accommodates failures of active reactivity control and active primary sodium heat removal systems. When these active systems are successful, sub-criticality is quickly reached and decay heat is accommodated without a challenging heat-up of the core or primary sodium coolant, meaning that stable plant conditions have been achieved. The success criteria analysis confirms that a mission time of 24 hours is appropriate for sequences where scram is successful and primary sodium heat removal is provided by BOP Cooling or Forced Air ACS (i.e., a backup heat removal system that operates by natural circulation air cooling of the Steam Generator outer shell while sodium is available in the Steam Generator).

Sequences where primary sodium heat removal must be provided by passive systems (RVACS, Passive ACS) are accommodated by the PRISM design, but stable plant conditions may not be reached within 24 hours for these scenarios since they develop slowly. When decay heat removal is accomplished by RVACS, a 72-hour extended mission time is appropriate to ensure that a safe and stable plant condition has been reached. This mission time has been confirmed by the success criteria analysis.

Sequences where reactivity control must be provided by passive inherent reactivity feedback (IRF) are accommodated by the PRISM design, but the reactor will not be subcritical unless control rods or Ultimate Shutdown System (USS) assemblies are inserted into the core. Previous calculations indicate that this must happen within days of the event. For the purpose of this analysis, a 24-hour mission time is assumed. The plant will be placed in a stable condition once either control rods or USS assemblies are inserted.

One of the features of the PRISM PRA model was the approach to determining success criteria for passive features, namely the air cooling by RVACS and the IRF features of the core that intrinsically suppresses reactivity when reactor power increases. The PRISM reactor core is designed to provide strong inherent negative reactivity feedbacks with rising temperature. With this characteristic and RVACS heat removal capability, the PRISM reactor is capable of safely withstanding severe undercooling and overpower transient events even with a failure to scram. As the temperature increases during an event, the negative feedbacks from the radial core expansion, grid plate expansion, axial core assembly expansion, Doppler, and control rod drive line expansion are activated, creating a net negative reactivity for the core. To evaluate the reliability of IRF, the key question is: how fast do the IRF mechanisms respond to an initiating event to reduce power? If the IRF mechanisms reduce power too slowly, fuel damage may not be prevented. Monte Carlo sampling calculations were performed to give probabilities of various levels of fuel damage being reached.

Innovative reliability models are employed to model the passive features of RVACS. The key question is: what levels of degradation would result in damage to fuel? To answer this question, the states of six major boundary conditions were evaluated for three cases: a normal condition, an off-normal condition, and an extreme condition. Calculations were run for each of the 729 combinations to determine the level of damage to the fuel within the vessel, if any. The PRISM systems and functional failure modes that could adversely affect key safety function performance are listed in Table 3-6. Table 3-7 defines the passive function categories.

Table 3-6. PRISM Systems and Functions Modeling

System Name	Postulated Failures Modeled	Safety Function	Passive / Active
Primary Heat Transport System	Loss of flow initiating event EM pumps fail to coastdown EM pump(s) fails to trip	Core Flow Core Flow Heat Removal	Active Passive Cat D Passive Cat D
Intermediate Heat Transport System	Failure of IHTS to move heat to ACS and BOP systems Intermediate EM pump(s) fail to trip	Heat Removal Heat Removal	Passive Cat B Passive Cat D
Steam Generator System / BOP	BOP SG heat removal unavailable for single module IEs BOP SG heat removal unavailable for power block IEs Failure to isolate module from main steam header Non-arrested SG/IHTS pressurization from a sodium-water reaction	Heat Removal Heat Removal Heat Removal Confinement & Heat Removal	Active Active Active Active
Active Reactivity Control Systems: - Control Rod System - RPS (Scram) - USS	Failure of Control Rod Drive system scram function USS assemblies fail to insert in core	Reactivity Control Reactivity Control	Active Active
Inherent Reactivity Feedback Mechanisms	Reactivity feedback insufficient to prevent fuel damage from overheating	Reactivity Control	Passive Cat A
RVACS	Passive decay heat removal degradation	Heat Removal	Passive Cat B
Auxiliary Cooling System	Passive heat removal through natural circulation around steam generator shell Active heat removal through forced flow around steam generator shell	Heat Removal Heat Removal	Passive Cat B Active
Electrical AC Electrical DC	Numerous loss of power supports to other systems	Supporting Supporting	Active Active
Confinement (Reactor Vessel & Containment)	Primary system radionuclide barrier faults Containment radionuclide barrier faults Bypass the containment boundary	Confinement Confinement Confinement	Active Active Active
Reactor Component Cooling Water System	Provide cooling to active supporting equipment (e.g., AC generator)	Supporting	Active
Chilled Water System	Provide room cooling to active supporting equipment (e.g., AC generator)	Supporting	Active
Condensate Storage and Transfer System	Provide makeup to condenser hotwell	Heat Removal	Active
Instrument Air	Provide air supply to valves	Supporting	Active
Condensate and Feedwater System	Provide Feedwater to SG	Heat Removal	Active
Main Condenser and Circulating Water System	Provide cooling to main condenser	Heat Removal	Active
Digital Instrumentation and Controls	Provide automatic actuation signals Provide interface and software conduit for manual actuation signals	Supporting Supporting	Active Active

Table 3-7. IAEA Passive Function Categories[67]

Category	Characteristics	Examples
A	No signal inputs of "intelligence" No external power sources or forces No moving mechanical parts No moving working fluid	Nuclear Fuel Cladding Primary Coolant Boundary Inherent Reactivity Feedback
B	No signal inputs of "intelligence" No external power sources or forces No moving mechanical parts Moving working fluids	Containment Cooling Systems based on natural circulation of air flowing around the containment walls
C	No signal inputs of "intelligence" No external power sources or forces Moving mechanical parts Whether or not moving working fluids are also present	Overpressure Protection and/or Emergency Cooling Devices of Pressure Boundary Systems based on fluid release through relief valves
D	Contain inputs, mechanical parts and working fluid, but meet the following criteria: (1) energy must only be obtained from stored sources such as batteries or compressed or elevated fluids; (2) active components are limited to controls instrumentation and valves (single-action relying on stored energy); (3) manual initiation is excluded	Emergency Core Cooling Systems, based on gravity driven flow of coolant, activated by valves which break open on demand

One of the objectives of the PRISM PRA described herein was to benchmark the ASME/ANS PRA Standard for Advanced non-LWRs. The entire development of the PRA models was guided by the requirements of the standard. Feedback from this benchmark is currently being considered by the Writing Group responsible for the standard in preparation of a revised ANSI PRA standard for non-LWR PRAs.

3.9.4 Development of Event Sequences

To model the PRISM SFR event sequences, three general groups of event trees are required:

- Protected
- Unprotected
- Confinement

The protected trees provide the logic for sequences in which active reactivity insertion is successful via the control rods. The unprotected trees accommodate those scenarios where control rod insertion fails so that inherent reactivity feedback and the ultimate shutdown system must satisfy the reactivity control safety function. Finally, the confinement trees analyze the various radionuclide barrier success and failure combinations that ultimately result in the event sequence end states. A simplified representation of this approach is shown in Figure 3-15 for the Loss of Primary Forced Flow IE.

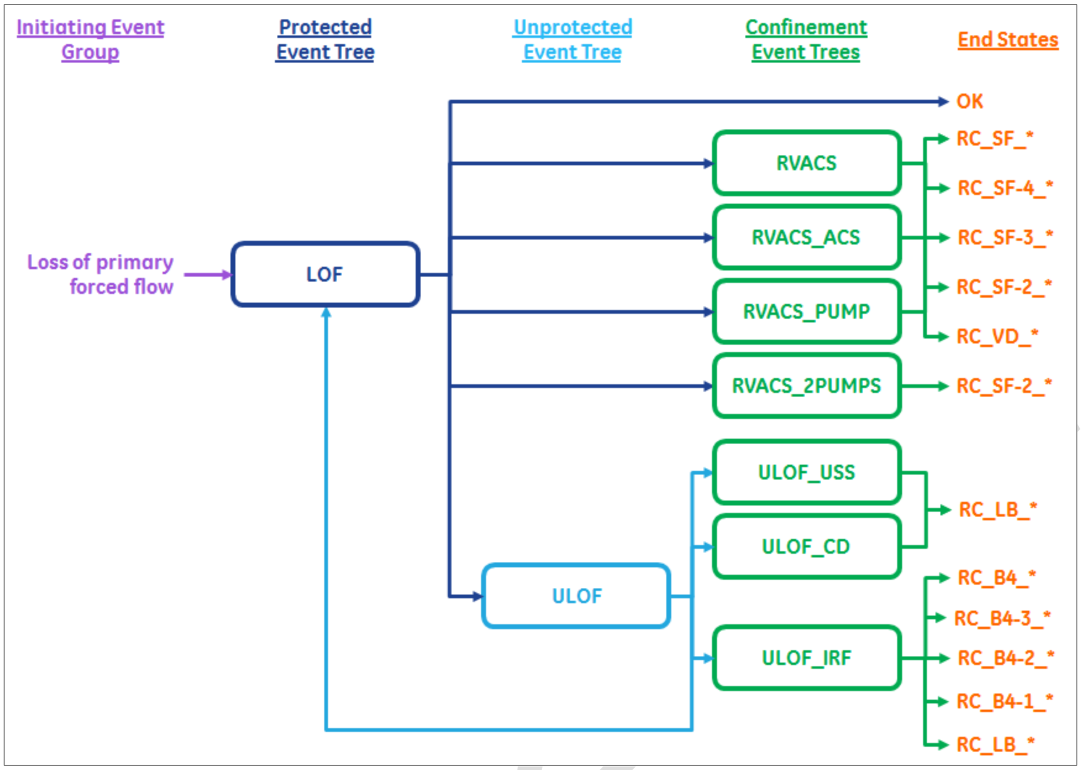


Figure 3-15. Example Event Sequence Overview Diagram

The protected and unprotected event trees identify the potential sequences that can lead to radionuclide release outside containment. Many of the sequences have common characteristics with respect to the challenge to the radionuclide barriers. These sequences are grouped using end states that are defined and analyzed within the Mechanistic Source Term analysis.

End states were selected to cover the range of possible plant conditions for the PRISM design, given the spectrum of IEs. End states take the form of either a safe, stable state (OK) or result in a radionuclide release outside of containment. The latter category of end states is referred to as release categories. All radionuclide releases are identified regardless of their magnitude.

The protected event trees provide the starting point for the event sequence analysis and are developed based on the IE groups identified in the IE chapter. The protected event trees developed for the PRISM PRA are as follows:

- BOP—BOP/loss of heat sink (LOHS) faults
- IHTS—IHTS large leak
- IHX—IHX bypass leak
- LOF—Loss of primary forced flow
- LOOP—Loss of offsite power
- NSSS—NSSS faults/general transients

- SGTR—Steam generator tube rupture
- TOP—Transient overpower

Each IE group is assigned to a protected event tree. Credit is taken for the systems and functions that are capable of responding to the event, as well as key operator actions that are available to mitigate the event. Event sequences in the protected event trees that do not lead to safe, stable conditions or release categories transfer to another protected, unprotected, or confinement event tree.

The unprotected event trees were developed to evaluate event sequences that involve a failure of the reactor to scram via the Control Rod Drive system. In this event, the inherent reactivity feedback mechanisms of Doppler broadening, changes in sodium density and metallic fuel expansion and contraction respond to the core temperature increases by bringing the core to a safe, and stable, near zero fission power state. As a backup, the USS can also shut down the reactor. Owing to the negative reactivity feedbacks, potentially severe, but extremely unlikely accidents can be accommodated with benign consequences. Three postulated, beyond design basis scenarios can illustrate this: inadvertent withdrawal of all control rods without scram (unprotected transient overpower), loss of primary pump power and loss of all cooling by the IHTS without scram (unprotected loss of flow/loss of cooling), and loss of all cooling by the IHTS without scram (unprotected loss of cooling).

Based on the three beyond design basis scenarios listed above, the following unprotected event trees are developed for the PRISM PRA:

- ULOF—Unprotected loss of primary forced flow
- ULOHS—Unprotected loss of heat sink
- UTOP—Unprotected transient overpower

The confinement event trees were developed to evaluate the release categories for protected and unprotected event sequences. The confinement event trees developed for the PRISM PRA include:

- RVACS—RVACS degradation with passive ACS success
- RVACS_PUMPS—RVACS degradation with pump heat from EM pumps
- RVACS_ACS—RVACS degradation with passive ACS failure
- ULOF_CD—ULOF with EM pump coastdown failures
- ULOF_IRF—ULOF and IRF/ gas expansion modules failure
- ULOF_USS—ULOF with failure to manually insert USS rods
- ULOHS_IRF—ULOHS and IRF failure
- LOHS_US—ULOHS with failure to manually insert USS rods

- UTOP—UTOP and IRF failure
- UTOP_USS—UTOP with failure to manually insert USS rods

The outcome of each event tree mitigating function (i.e., success or failure) is determined by success criteria analysis described earlier. Success criteria analysis defines the mitigating system and functions that are included in the PRA model, including combinations of systems and functions that prevent radioactive release. Each event tree node and its respective mitigating function are designated as event tree top events. The SSC supporting mitigation functions are:

- Forced air ACS
- Passive ACS
- BOP cooling
- Confinement integrity
- Containment isolation
- Reactivity control of control rods
- Steam generator sodium water reaction arrest (relief-isolation-blowdown)
- Fuel barrier damage state
- IHTS available/natural circulation
- Tripping of intermediate EM pumps
- Pump coastdown
- Tripping of primary EM pumps
- IRF
- Gas expansion modules
- RVACS cooling
- USS reactivity control
- Vessel head integrity

3.9.5 Event Sequence End States

Release categories are defined for the event sequence end states based on similar release characteristics. Initially, all event sequences resulting in radionuclide release are identified. The different characteristics of these releases are summarized. Finally, the end states are grouped into formal release categories.

For the PRISM plant design, the characteristics of radionuclide release are studied for all instances where a sequence results in damage to the cladding or fuel within the reactor vessel. For completeness, this accounting includes sequences where the reactor vessel head and containment are successfully performing their confinement functions as radionuclide barriers. In this way, the very small release that results from clad damage and the assumed level of leakage from the vessel and containment are quantified for PRISM for any level of clad damage.

The radionuclide release categories that group the release event sequences have several common features and differentiating attributes. Postulated radionuclide releases from the PRISM design take the following release path. Fuel cladding fails primarily by hoop stress rupture and releases fission products to the sodium coolant. Many isotopes remain trapped within the fuel matrix and sodium and do not contribute to the source term. The radionuclides that escape the sodium by vaporization or boiling are released to the cover gas space between the sodium hot pool surface and the vessel head. From the cover gas space, the radionuclides either leak out through failed seals in the vessel head, or if seals are not failed, gases leak at an assumed rated design leakage rate. Gases that escape the cover space by failure or by leakage end up in the upper containment. Any radionuclide release to the environment from the gases that reach upper containment is by either a failure of the containment isolation function or by rated design leakage if containment isolation is successful. All source terms that take this path are ground level releases, with no credit for retention in the reactor building because the upper containment resides near ground level.

The exception to the release path described above is in sequences where bypass of the vessel head and containment barriers is postulated following a postulated rupture of IHX tubes. Realistically, this is not believed to be a direct release path to the environment, given that the static head of the IHTS would keep sodium flowing from the intermediate loop to the primary loop. However, no mechanistic modeling of radionuclide release from a damaged IHX through the IHTS has been performed. Therefore, any damage to IHX tubes is assumed to result in a release that bypasses containment.

The time of barrier failure is driven by the thermal-hydraulic calculations of fuel cladding performance in various sequences. After clad failure, no additional delay time is considered for transport of radionuclides through the hot pool to the cover gas space. The transport of radionuclides from the cover gas space to the containment is either controlled by the design leak rate through the vessel head, or the transport is assumed to be instantaneous if there is a failure of the vessel head seals. Similarly, transport from the containment to the environment is either controlled by the leak rate or is considered simultaneous given a failure of containment isolation.

Table 3-8 is an excerpt from the Mechanistic Source Term Analysis (mentioned in Table 4-2 in Reference [43]) which shows some release categories. Each category identified contains a listing in the table of all intact and not intact radionuclide release barriers and each release is described. Release barriers include the combinations of the following faulted or intact states: metal fuel matrix, fuel cladding, hot pool sodium, vessel head, and containment.

Table 3-8. Release Category Definitions[43]

Release Category	Release Barriers Intact (✔) or Not Intact (✘)	Radionuclide Release Description
RC_B4_1	✔ All metal fuel matrices ✔ Clad in spent fuel & batches 1-3 ✘ Clad in batch 4 ✔ Hot pool sodium ✔ Vessel head ✔ Containment	Radionuclides within cladding of every pin in core batch 4 are released to sodium coolant
RC_B4_2	✔ All metal fuel matrices ✔ Clad in spent fuel & batches 1-3 ✘ Clad in batch 4 ✔ Hot pool sodium ✔ Vessel head ✘ Containment	Release is dominated by inventory from these pins that is not retained by the sodium coolant and escapes the hot pool to the cover gas space (i.e. noble gases)

3.9.6 LBEs as Event Sequence Families

Selecting LBEs for the PRISM PRA followed the same process that is described in the previous example. Event sequence families are used to group together two or more event sequences when the sequences have a common IE, safety function response and end-state. 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. Event sequences with similar but not identical safety function responses are not combined when such combinations would mask the definition of unique challenges to the SSCs that perform safety functions.

3.9.7 Example PRISM Event Sequence for Loss of Flow Event

The purpose of this section is to present an example of the event sequence development for an IE that is common to SFR designs. This example includes a definition of the IE, the development of an event sequence, and an event tree which quantifies the event sequence frequencies, assignment of end states, and a classification into LBE categories.

3.9.7.1 Definition of Initiating Event

The IE for this example is Loss of Primary Flow, which considers total or partial loss of forced flow in the primary sodium loop. It centers around the possible faults of the primary EM pumps to function.

3.9.7.2 Safety Design Mitigation Strategy

EM pumps have no inertia, so to prevent flow stagnation in the core region following EM pump stoppage, synchronous machines provide an artificial coastdown. These coastdown machines are typically flywheels coupled with motor-generator units. They are operated continuously so that there will be a coastdown if there is a power loss or other faults causing the primary EM pumps to trip. As the synchronous machines coast down, the rotational energy is converted to electrical power for the primary EM pumps. They experience a gradual reduction in pumping power, and thus provide better removal of the relatively high decay heat immediately following a scram or a passive shutdown.

3.9.7.3 Key Assumptions

The following assumptions are made to support the loss of flow event sequence development for the advanced conceptual design and will be replaced by appropriate analyses in the final design phase.

- The primary and intermediate loop EM pumps are assumed to release their entire rated gross thermal power to the sodium if their breakers fail to open.
- It is assumed that local boiling always leads to fuel damage. This assumption is made in lieu of detailed analyses of fuel, neutronics and thermal-hydraulic relationships for stagnated flows.
- For long-term decay heat transients, the Intermediate Heat Transfer System is assumed to provide the necessary heat transfer via natural circulation in the Steam Generator. Any failure probabilities associated with this natural circulation are assumed to be negligible when compared to other heat transfer failure mechanisms that are modeled.

3.9.7.4 Event Sequence Development

The event tree for Loss of Flow events is shown in Figure 3-16. With a loss of primary forced flow, a scram signal is generated on power-to-flow mismatch. Failure to insert control rods transfers to the Unprotected Loss of Flow event tree which considers coastdown of the EM pumps and power reduction due to negative reactivity feedbacks. Successful control rod insertion takes the sequence to decay heat removal functions. If the preferred heat sink (Main Condenser/Balance of Plant) is available, the reactor is brought to a safe stable shutdown. If it is unable to provide heat removal, then Forced Air Auxiliary Cooling can be actuated. This system removes heat from the exterior of the shrouded Steam Generator by an induction fan, and relies on a natural circulation flow path in the intermediate loop.

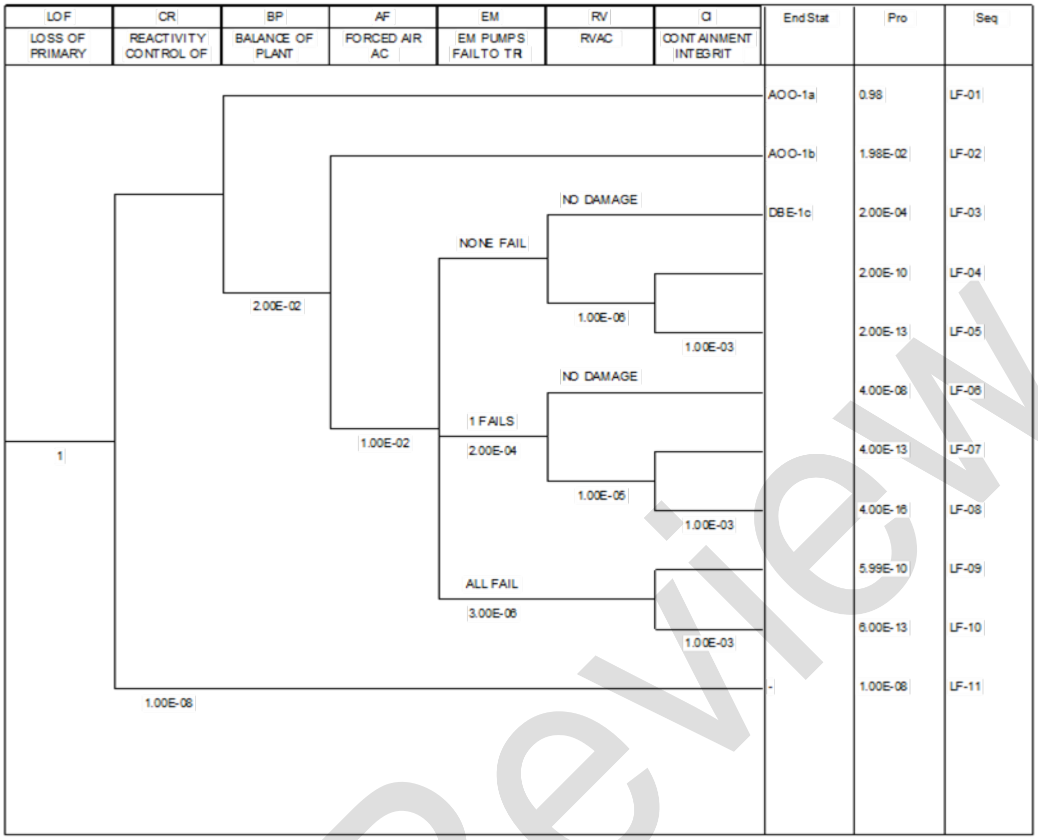


Figure 3-16. Simplified Event Tree for PRISM Loss of Flow Event

The residual heat from EM pumps can be a significant load if the EM pump trip breakers fail to open. Thus, three outcomes are modeled. RVACS passively removes heat from the vessel to the atmosphere by establishing a natural convection flow path for air to circulate in and around the vessel/lower containment and to exit back to the atmosphere. RVACS is sized to remove sufficient heat to prevent primary sodium temperature from causing cladding failure and fuel damage to the spent fuel in the in-vessel racks and core. Because RVACS is a passive system, its performance is subject to degradation. Thus, to account for uncertainties an RVACS failure (degradation) branch is applied to the tree. It is actually there to account for the possibility of degraded heat removal capability due to external factors in which RVACS cannot remove sufficient heat to prevent further heating of the core and sodium. Thus far, all end states are safe stable shutdowns. However, with a postulated BOP/ACS/RVACS fault chain, spent fuel damage is assumed to eventually occur, albeit in a few days. Therefore, the Reactor Vessel and Containment barrier effects on the source term from the core/sodium are challenged and the final end states are thus identified as either AOOs, DBEs, or BDBEs.

Quantification of this event tree was made using engineering judgments and assumptions that will be replaced with appropriate design detail and analysis when the design is finalized. However, these preliminary estimates are used to yield a preliminary set of LBEs, whose results are listed in Table 3-9.

Table 3-9. LBEs for a Simplified Loss of Flow Event

LBE No.	LBE Type	Plant Response	Frequency per Plant-Year
LF-01	AOO	Reactor Scram, Balance of Plant cooling removes decay heat; Safe Shutdown	0.98
LF-02	AOO	Reactor Scram, Forced Air Aux Cooling removes decay heat; Safe Shutdown	1.98 E-2
LF-03	DBE	Reactor Scram, RVACS removes decay heat; Safe Shutdown	2.0 E-4
LF-04	NR	Reactor Scram, Degraded decay heat removal from RVACS, core damage with nominal leakage past reactor vessel head and nominal leakage past containment	< E-8
LF-05	NR	Reactor Scram, Degraded decay heat removal from RVACS, core damage with nominal leakage past reactor vessel head and penetration leakage or bypass past containment	< E-8
LF-06	NR	Reactor Scram, residual heat from one untripped EM pump, RVACS removes pump and decay heat; Safe Shutdown	4 E-8
LF-07	NR	Reactor Scram, Degraded decay heat removal from RVACS, residual heat from one untripped EM pump, core damage with penetration leakage past reactor vessel head and nominal leakage past containment	< E-8
LF-08	NR	Reactor Scram, Degraded decay heat removal from RVACS, residual heat from one untripped EM pump, core damage with penetration leakage past reactor vessel head and penetration leakage or bypass past containment.	< E-8
LF-09	NR	Reactor Scram, all EM pump breakers fail to trip, core damage with penetration leakage past reactor vessel head and nominal leakage past containment	< E-8
LF-10	NR	Reactor Scram, all EM pump breakers fail to trip, core damage with penetration leakage past reactor vessel head and penetration leakage or bypass past containment	< E-8
LF-11	NR	Unprotected Loss of Flow (failure to scram); Transfer to ULOF event tree	1 E-8

Note: NR = No rating as the BDBE category is defined as between 1E-4 and 5E-7/plant-year

3.9.8 Example PRISM PRA Results

Example results evaluated for the PRISM reactor are shown in Figure 3-17. Sequences with RVACS success are AOOs and reside at the y-axis (zero consequence). With RVACS degradation, DBEs and BDBEs are possible; however, these very preliminary results only appear in the lower uncertainty range with a frequency greater than E-8 per year. Site exclusion area boundary doses are compared to the TLRC proposed limits for acceptable frequencies and consequences as proposed in the LBE selection white paper. The site boundary dose consequences are based on mechanistic source terms that were evaluated for PRISM event sequences grouped into LBE categories. The development of LBEs from the PRISM PRA results is described in Reference [2].

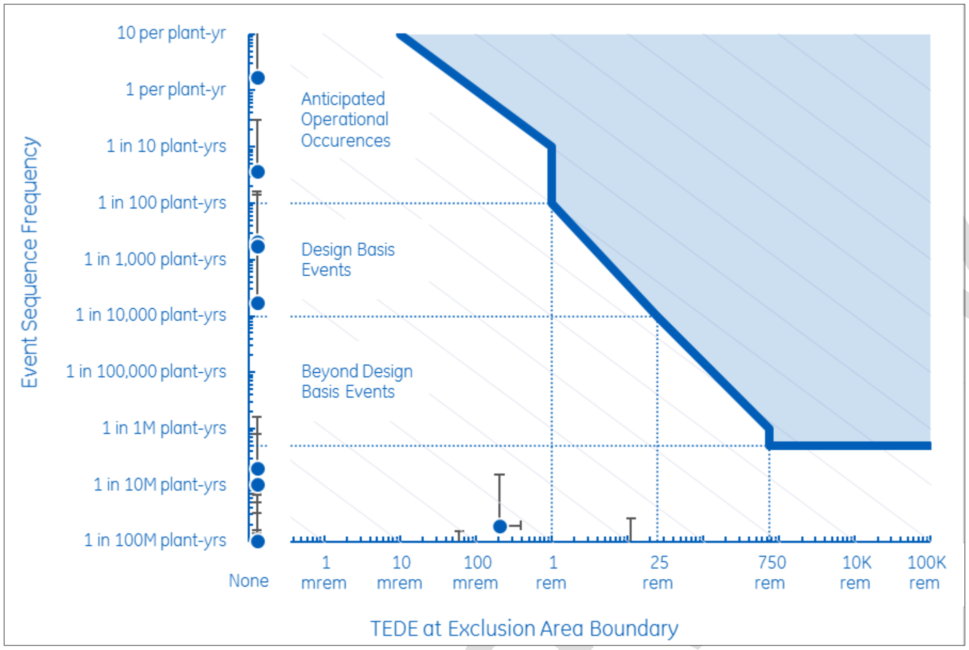


Figure 3-17. Example PRISM LBEs Compared with TLRC Frequency-Dose Evaluation Criteria

4.0 ADVANCED NON-LWR PRA TECHNICAL ADEQUACY FOR RISK-INFORMED AND PERFORMANCE-BASED APPLICATIONS

The purpose of this section is to discuss technical issues and challenges for advanced non-LWR PRA and the proposed approach for achieving technical adequacy that is fit-for-purpose for the intended PRA applications during design development and licensing.

4.1 Technical Issues and Challenges for Advanced non-LWR PRA

There are several technical issues and challenges in developing the advanced non-LWR PRAs within the LMP framework. Providing a context for the approach to PRA technical adequacy requires resolution of these issues, which are listed below and are discussed in the following sections.

- PRA treatment of multi-reactor module plants
- Sufficiency of relevant PRA data
- Treatment of inherent and passive safety features
- New risk-informed applications for non-LWR PRAs

4.1.1 PRA Treatment of Multi-Reactor Module Plants

As discussed more fully in the companion paper on the LMP approach to selection of LBEs,[2] the scope of the PRA is intended to cover all the reactor modules and radionuclide sources that may be within the scope of an advanced non-LWR plant design. This is to ensure that there is sufficient feedback to the plant design team on multi-module and multi-source risk insights so that design strategies to manage the risks of multi-module and multi-source accidents can be implemented. An important safety design strategy is to ensure that there are no risk significant accident sequences that involve releases from two or more reactor modules or radionuclide sources. This strategy is also required to justify the definition of design basis accidents in a manner that single reactor source terms are involved.

Although the vast amount of experience with the performance of nuclear reactor PRAs has been limited to single reactor PRAs, there have been multi-unit and multi-module PRAs on LWRs as well as non-LWRs. In addition, there are number of references available to provide useful guidance in the performance of both multi-unit PRAs for operating LWR plants as well as future non-LWR plants that are based on a modular plant design. A summary of some key available references is provided in Table 4-1. The example PRAs summarized in the previous section for the MHTGR and PRISM both included multi-module treatment. Examples of LBEs developed from these PRAs that involve both single reactor units, and multiple reactor units are provided in the LMP white paper on LBE selection and evaluation in Reference [2]. However, neither of these PRAs included non-core sources of radioactive material due to lack of sufficient design details on the systems and structures associated with these sources.

Table 4-1. References for Multi-Module PRA Development

Category	Reference
Non-LWR Case Studies	MHTGR PRA (4 Reactor Modules)[4] PRISM PRA (2 Reactor Modules)[40] HTR-PM PRA (2 Reactor Modules)[60]
Non-LWR Guidance and Standards	ASME/ANS Non-LWR PRA Standard[42] NGNP PRA White Paper[1]
LWR Case Studies	Seabrook PRA (2 Reactor Units)[50] NRC Level 3 PRA (2 Reactor Units)[53] NuScale Multi-module PRA[71]
LWR Guidance and Standards	IAEA Technical Approach to MUPSA[49][47] IAEA TECDOC 1804[48] Canadian Nuclear Safety Commission International Workshop on MUPSA[51]

Frameworks for performing a multi-unit or multi-module PRA have been developed in several of these references. The framework developed by the IAEA as part of a Technical Approach for Multi-Unit Probabilistic Safety Assessment (PSA)[49] is shown in Figure 4-1. The approach followed in the NRC Level 3 PRA project[53] is shown in Figure 4-2. The approach followed in both of these cases is to first perform a single reactor unit PRA, and then to use the data and models from the single reactor PRA to build a model that addresses accidents involving two or more reactor units or modules.

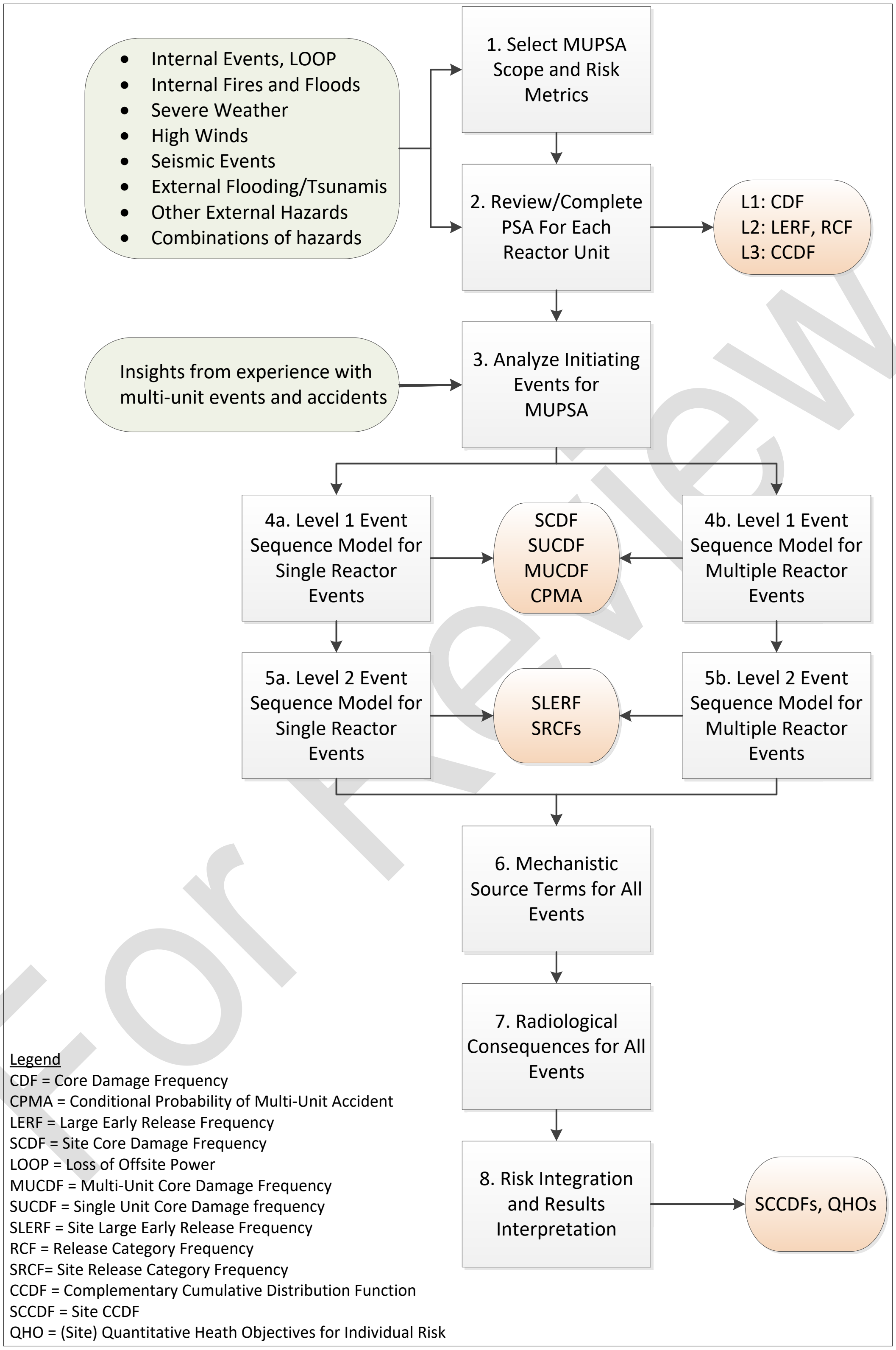

Internal Events, LOOP
Internal Fires and Floods
Severe Weather
High Winds
Seismic Events
External Flooding/Tsunamis
Other External Hazards
Combinations of hazards
1. Select MUPSA Scope and Risk Metrics
2. Review/Complete PSA For Each Reactor Unit
L1: CDF
L2: LERF, RCF
L3: CCDF
Insights from experience with multi-unit events and accidents
3. Analyze Initiating Events for MUPSA
4a. Level 1 Event Sequence Model for Single Reactor Events
SCDF
SUCDF
MUCDF
CPMA
4b. Level 1 Event Sequence Model for Multiple Reactor Events
5a. Level 2 Event Sequence Model for Single Reactor Events
SLERF
SRCFs
5b. Level 2 Event Sequence Model for Multiple Reactor Events
6. Mechanistic Source Terms for All Events
7. Radiological Consequences for All Events
8. Risk Integration and Results Interpretation
SCCDFs, QHOs
Legend
CDF = Core Damage Frequency
CPMA = Conditional Probability of Multi-Unit Accident
LERF = Large Early Release Frequency
SCDF = Site Core Damage Frequency
LOOP = Loss of Offsite Power
MUCDF = Multi-Unit Core Damage Frequency
SUCDF = Single Unit Core Damage frequency
SLERF = Site Large Early Release Frequency
RCF = Release Category Frequency
SRCF= Site Release Category Frequency
CCDF = Complementary Cumulative Distribution Function
SCCDF = Site CCDF
QHO = (Site) Quantitative Heath Objectives for Individual Risk

Figure 4-1. IAEA Framework for Multi-Unit Site PRAs[49]

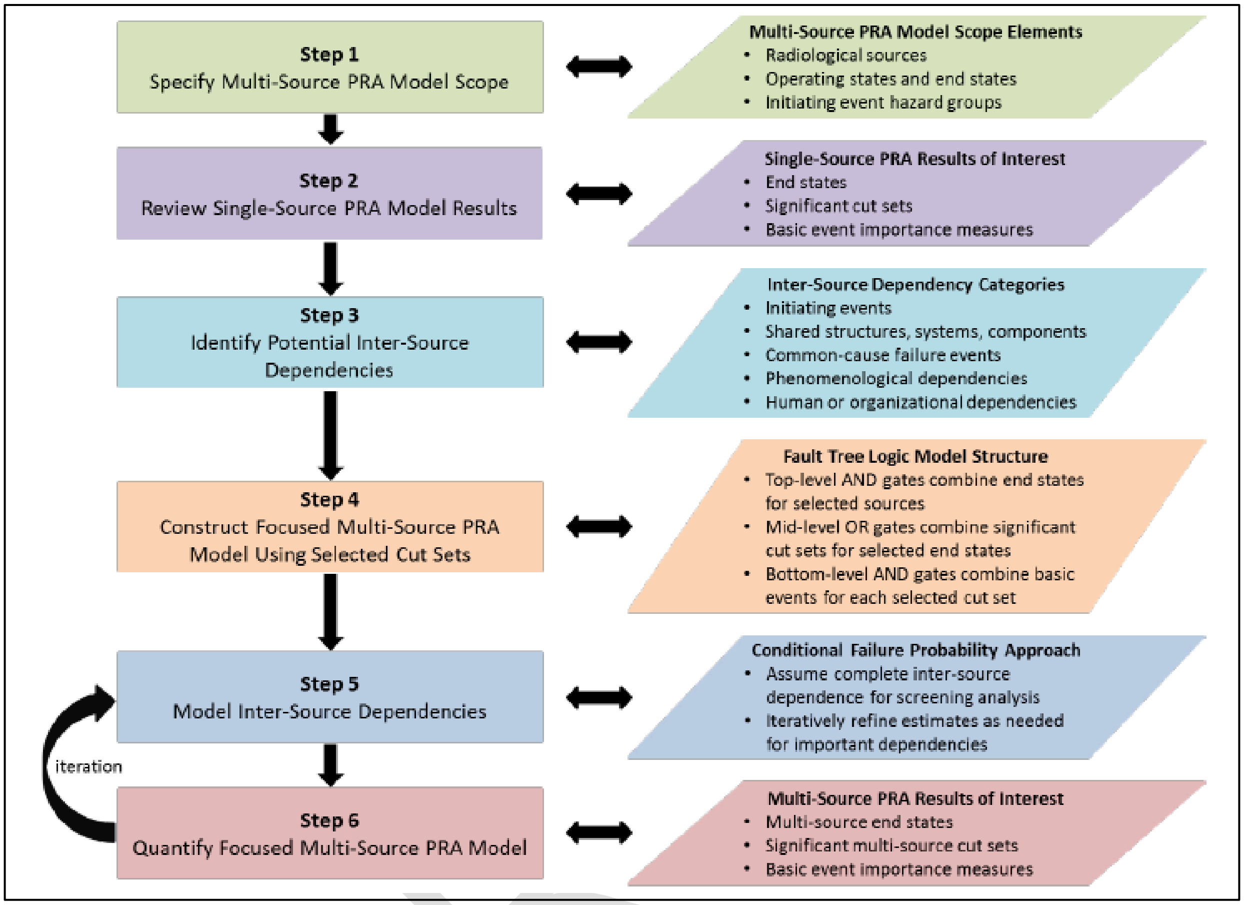

Step 1
Specify Multi-Source PRA Model Scope
Multi-Source PRA Model Scope Elements
• Radiological sources
• Operating states and end states
• Initiating event hazard groups
Step 2
Review Single-Source PRA Model Results
Single-Source PRA Results of Interest
• End states
• Significant cut sets
• Basic event importance measures
Step 3
Identify Potential Inter-Source Dependencies
Inter-Source Dependency Categories
• Initiating events
• Shared structures, systems, components
• Common-cause failure events
• Phenomenological dependencies
• Human or organizational dependencies
Step 4
Construct Focused Multi-Source PRA Model Using Selected Cut Sets
Fault Tree Logic Model Structure
• Top-level AND gates combine end states for selected sources
• Mid-level OR gates combine significant cut sets for selected end states
• Bottom-level AND gates combine basic events for each selected cut set
Step 5
Model Inter-Source Dependencies
Conditional Failure Probability Approach
• Assume complete inter-source dependence for screening analysis
• Iteratively refine estimates as needed for important dependencies
iteration
Step 6
Quantify Focused Multi-Source PRA Model
Multi-Source PRA Results of Interest
• Multi-source end states
• Significant multi-source cut sets
• Basic event importance measures

Figure 4-2. Multi-Unit Risk Approach in NRC Level 3 Project[53]

In this part of the model, the IEs from the single reactor model need to be analyzed to break down which events affect a single module and those that impact two or more modules. Multi-module IEs would typically include loss of offsite power, loss of any shared systems, Internal hazard events such as fires and floods that may occur in any shared structures, and most of the IEs caused by an external hazard such as seismic events and external flooding. Some of the technical issues that must be addressed in the modeling of multi-module accidents include:

- Increased stress and workload on plant operators in implementing emergency operating procedures, accident management, and emergency planning
- Analysis of common cause failures to distinguish between those confined to a single module and those that may occur in different modules
- Correlation of seismic fragilities for SSCs appearing in single and multiple modules characterized by the same fragility curve
- Treatment of time dependent releases that may occur from more than one location in the plant
- PRA modeling complexity when the plant to be analyzed includes more than two reactor modules

- Need to select risk metrics that sufficiently capture the dimensions of multi-unit risk including the increase in the frequency of a single module accident due to more modules being at risk and the introduction of new types of accidents that involve two or more reactors or radionuclide sources

These and other issues have been addressed in the available case studies and guidance documents to varying degrees. It is emphasized that that inclusion of multiple modules and sources serves to capture sufficient risk insights to implement effective risk management strategies.

4.1.2 Sufficiency of Relevant PRA Data

Questions have been raised as to whether there are sufficient PRA data to perform advanced non-LWR PRAs and whether uncertainties associated with availability of relevant data can be sufficiently addressed.

The PRA data parameters for the non-LWR PRA database will include the following data categories:

1. Failure rates and unavailability parameters for active components unique to the non-LWR (e.g. gas blowers and compressors for HTGRs and electromagnetic pumps for liquid metal reactors)
2. Failure rates and unavailability parameters for active components common to LWRs (e.g., pumps and valves in water systems, water-to-water heat exchangers, diesel generators, breakers, and instrumentation and control components)
3. Common cause failure parameters for a limited set of redundant components (Based on experience with the MHTGR and PRISM PRAs, these parameters are mostly in common cause groups of components typical for LWRs.)
4. IE frequencies and failure probabilities for passive component failure modes (e.g., pipes, pressure vessels, weldments, and pressure relief valves)
5. IE frequencies for power conversion system and other equipment failure modes common to LWRs (e.g. loss of feedwater, turbine trip, loss of offsite power, electrical system faults)
6. IE frequencies for internal and external plant hazards found in full-scope LWR PRAs (fires, floods, seismic events, transportation accidents)

Of the six categories of data parameters listed above, the ones that are subjected to the most uncertainty due to lack of relevant operating experience for non-LWRs in general are Categories 1 and 4 and to a lesser extent Category 3. For the remaining categories, as well as most of the parameters expected in Category 3, the advanced non-LWR may benefit from PRA data that have been developed for LWRs. The extent of non-LWR PRA data development, of course, will vary among the various reactor types. There is a rich history of PRA development for SFRs and HTGRs. The database that was developed to support the recent PRA on PRISM, which benefits from operating experience with sodium reactors is summarized in Reference [61]. The PRA data developed for the MHTGR PRA, which benefited from service experience with gas-cooled reactors in the U.K, is documented in Reference [4]. For the PBMR project in South Africa, service experience with LWRs and evaluations of HTGR degradation mechanism were used to develop HPB IE frequencies as illustrated in Figure 4-3. This assessment benefitted from the use of LWR materials and design codes for the HPB piping.

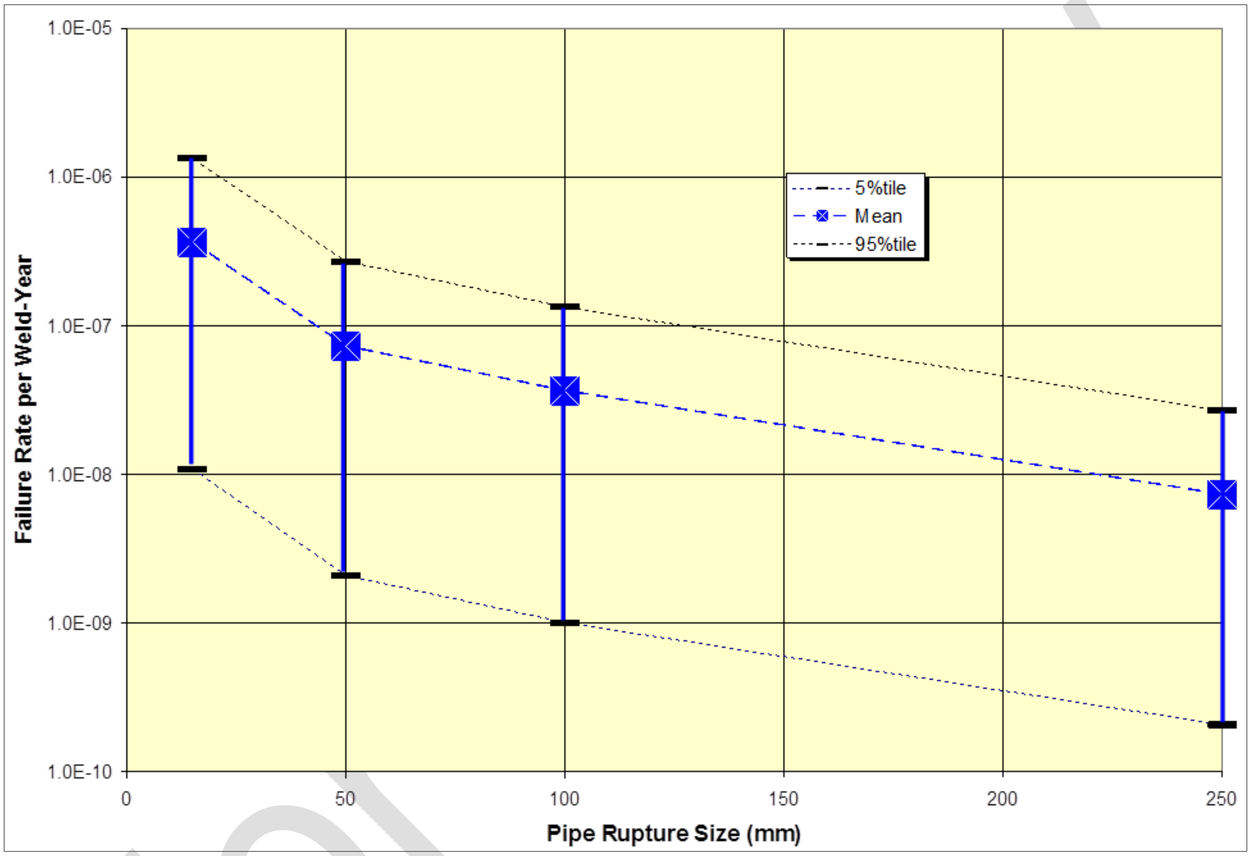


Figure 4-3. Failure Rate vs. Rupture Size for 250 mm Carbon Steel Pipe Weld on PBMR HPB[1]

To provide a perspective on the challenges facing the non-LWR PRA database development, the fact that many of the PRA data parameters can be addressed using PRA data from LWR PRAs means that the challenge is not nearly as significant as it was for the first LWR PRA published in WASH-1400.[64] The Reactor Safety Study was performed before any of the then limited amount of service experience* was analyzed to develop failure rates and IE frequencies. Generic non-nuclear sources were used to develop component failure rate estimates. Loss of coolant accident frequencies were estimated using gas pipeline data and service experience from fossil fueled power plants and engineering judgements to account for the expected reliability improvements from ASME nuclear piping and pressure vessel codes. Today, with the benefit of

* When the Reactor Safety Study was performed, there were only about 400 reactor-years of service experience with LWRs, and no studies had been performed to convert this experience into PRA data parameters.

extensive service experience, uncertainties in the values of data parameters for relatively frequent events have been greatly reduced. However, uncertainties in the values of data parameters for rare events continue to be large as even thousands of reactor years of service data is insufficient alone to provide a statistical basis for failure rate estimates. It is noted that today, LOCA frequencies are not estimated directly from data but rather from an expert elicitation.[65]

The PRA database developed in WASH-1400[64] was the primary basis for the first decade of PRA development until LWR service experience could be factored in. In retrospect, that data was sufficient to support the early applications of PRA which included the resolution of generic and unresolved safety issues associated with severe accidents. Advanced non-LWR PRA will fortunately need to deal with a smaller scope of data parameters that are not supported by available data. Notwithstanding, it will be important that uncertainties in the data parameter estimates be well characterized and managed in a fashion that reflects the state of knowledge regarding the parameters.

The NRC guidance on PRA uncertainty treatment in NUREG-1855[66] is sufficiently technology-inclusive to be useful for non-LWR PRAs. For some parameters, it may be necessary to perform an expert elicitation such as been the basis for current LWR LOCA frequencies.

4.1.3 Treatment of Inherent and Passive Safety Features

Evolutionary LWRs and advanced non-LWRs rely more on inherent and passive safety features and less on active SSCs in the performance of safety functions, consistent with the Commissions Advanced Reactor Policy.[24] Consistent with the approach that is used for the treatment of passive and inherent safety features, the approach used in non-LWR PRAs is to use phenomenological models supported by a robust uncertainty analysis to analyze the plant transient response to events, success criteria, and mechanistic source terms. This can be contrasted with traditional PRA modeling of active systems based on fault trees, failure rates, maintenance unavailability parameters, common cause models and parameters, and human reliability models. Useful guidance on this aspect of the PRA is provided in the Standard Review Plan Chapter 19.[26] In the recent upgrade to the PRISM PRA, a comprehensive uncertainty analysis was performed to support the PRA success criteria and meet associated requirements for passive systems in the ASME/ANS PRA Standard for Advanced Non-LWRs as discussed more fully in Reference [63].

4.1.4 Need to Address New Risk-Informed Applications

Current industry and regulatory experience with risk-informed decision making has been with introducing incrementally changes to the licensing bases for currently operating LWR plants using a decision making process based on Regulatory Guide 1.174.[28] The technical adequacy of the PRA inputs to these risk-informed decisions has been established in Regulatory Guide 1.200[29] which endorses the available industry PRA standards. However, the technical adequacy of the risk-informed decisions is ultimately confirmed by NRC review of license amendment requests which are subject to full NRC review resulting in an amended license. This model makes sense given that the original licenses are based solely on meeting deterministic requirements. For new reactors, a summary of the required supporting PRA results is currently provided in Chapter 19 of the Design Certification Application or Combined License application, which is subjected to a full license review and supported by NRC audits of the supporting PRA documentation.

The PRAs performed under the LMP framework are intended for a broader range of risk-informed and performance-based decisions including input to selection of LBEs, SSC safety classification, and evaluation of DID adequacy. The approach being proposed to establish PRA technical adequacy is based on the use of the ASME/ANS PRA Standard for Advanced non-LWRs[43] to support the PRA development, which includes the necessary documentation and peer reviews. NRC staff participation on these peer reviews is recommended so that subsequent risk-informed decisions supported by the PRA are fully vetted. In addition, any risk-informed licensing decisions that are supported by the PRA would be subject to full licensing review by including the justification for the decision as part of the license application or supporting topical report. While the current trial use standard ASME/ANS-RA-S-1.4-2013[43] did not benefit from a final ballot review and endorsement, it would benefit the success of the LMP if NRC were to take an active role in the development of the revised non-LWR standard which is being prepared for a ballot for ANSI status in 2018. Such an active role would include continued participation on the Writing Group responsible for the standard, providing technical comments on the current trial use standard, and endorsing the revised standard when it has successfully completed the ballot. More information on the non-LWR Standard is provided in the next section.

4.2 Guidance and Standards for PRA Technical Adequacy

4.2.1 ASME/ANS PRA Standard for Advanced Non-LWR PRA RA-S-1.4-2013[43]

When the NGNP PRA white paper[1] was developed in 2011, the only published PRA standards were for currently operating LWR nuclear power plants. In December 2013, the ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM) issued a trial use PRA Standard for Advanced non-LWR Nuclear Power Plants.[43] This standard was developed to support PRA development and risk-informed applications on advanced non-LWR nuclear power plants. The stakeholders who participated in the development of this standard included the Exelon PBMR project, the DOE NGNP project, the China HTR-PM project, several SFR projects including GE-PRISM, the Argonne National Laboratory/Korea Atomic Energy Research Institute fast reactor project, and the TerraPower Traveling Wave Reactor project. Since the trial use standard was issued, representation on the JCNRM non-LWR project team responsible for the standard was expanded to include representation by the molten salt reactor community and the X-energy project to develop a small modular HTGR based on the pebble bed fuel concept.

Several representatives of the NRC participated on the JCNRM non-LWR project team responsible for developing the trial use standard but the agency has not endorsed it because there were no active licensing interactions with a prospective non-LWR licensee when the standard was issued for trial use. During the trial use period, which ended in December 2016, there were several pilot PRA projects that utilized the vast majority of the technical requirements in the standard and provided feedback to the project team which will provide the technical basis for a revised ANSI standard which is currently being developed. The following pilot PRAs were performed that provided feedback to the project team:

- GE-Hitachi performed a project for DOE which included a major PRA upgrade for the GE-PRISM reactor, a pool type liquid metal fast reactor. One of the objectives of this project was to pilot the non-LWR PRA standard. Public domain references for this PRA are found in References [40], [61], [62], and [63].
- A PRA was performed using the non-LWR standard to meet licensing requirements for the HTR-PM under construction in China. This reactor is a pebble bed type HTGR. A preliminary PRA was performed and included in the Preliminary Safety Analysis Report which was required to obtain a construction permit, and a more comprehensive PRA is currently being completed to meet a requirement for an operating license.[60]
- TerraPower is performing a PRA using the non-LWR standard to support the design of the Traveling Wave Reactor, a sodium-cooled fast reactor that is designed to utilize spent LWR fuel as a fuel source. Feedback from this PRA was incorporated into the trial use standard and continues to support the development of the next edition of the standard.
- Argonne National Laboratory has participated in the development of the trial use standard and has incorporated experience in supporting the design of another liquid metal fast reactor being developed in Korea. ANL has also participated in the GE-PRISM PRA upgrade and has used the requirements in the standard for mechanistic source terms to guide the development of source term technology for SFRs.
- The trial use standard was sponsored in part by the PBMR project in South Africa and the DOE NGNP project and reflected the lessons learned from those PRA projects.
- PRAs using the standard have been recently initiated for the X-energy pebble bed reactor, the Molten Chloride Fast Reactor, as well as HTGRs and sodium reactors under development in Japan.

At the September 2016 meeting of the JCNRM, it was decided to move forward with the development of an ANSI standard for non-LWR PRAs. The schedule to prepare the draft for ballot review is December 2018. By the time Phase 2 of this LMP is completed, it is quite likely that the next edition of this standard will be available to support future non-LWR PRAs. Hopefully, circumstances will permit NRC's continued involvement in the JCNRM project team responsible for this standard and for NRC to endorse this standard as it has endorsed the LWR PRA standards in RG 1.200.[29]

To provide a suitable PRA standard for HTGRs, SFRs, and a wide variety of prospective advanced reactor concepts, a reactor technology neutral approach to writing the PRA requirements was adopted. Approximately 80% of the technical requirements in this non-LWR standard are the same as or similar to PRA requirements in LWR PRA standards including ASME/ANS-RA-Sb-2013,[42] and the ANS PRA standards that have been and are being developed for low power and shutdown PRAs, and Level 2 and Level 3 PRAs. The requirements in the non-LWR PRA standard have been organized into 18 PRA elements, which are defined in Table 4-2. This table identifies similarities and differences with corresponding LWR PRA standards.

Table 4-2. Elements of ASME/ANS Non-LWR PRA Standard and Comparison with LWR PRA Standards

ASME/ANS-RA-S-1.4-2013[43]	Corresponding LWR PRA Standard
Plant Operating State Analysis (POS)	Similar to POS in ANS Low Power and Shutdown PRA standard[44] to support PRA models covering operating and shutdown modes
Initiating Event Analysis (IE)	Similar to IE in ASME/ANS-RA-Sb-2013[42] except that LWR IE categories are replaced by reactor technology neutral categories and both single unit and multi-unit initiators are included
Event Sequence Analysis (ES)	Similar to AS in ASME/ANS-RA-Sb-2013 except that event sequences are developed to user defined intermediate end states and release categories
Success Criteria Development (SC)	Similar to SC in ASME/ANS-RA-Sb-2013 except that safe stable end states are defined to prevent user defined end states rather than to prevent core damage and large early release
Systems Analysis (SY)	Similar to SY in ASME/ANS-RA-Sb-2013
Human Reliability Analysis (HR)	Similar to HR in ASME/ANS-RA-Sb-2013
Data Analysis (DA)	Similar to DA in ASME/ANS-RA-Sb-2013
Internal Flood PRA (FL)	Similar to FL in ASME/ANS-RA-Sb-2013
Internal Fire PRA (FI)	Similar to FI in ASME/ANS-RA-Sb-2013
Seismic PRA (S)	Similar to S in ASME/ANS-RA-Sb-2013
Other Hazards Screening Analysis (EXT)	Similar to EXT in ASME/ANS-RA-Sb-2013
High Winds PRA (W)	Similar to W in ASME/ANS-RA-Sb-2013
External Flooding PRA (XF)	Similar to XF in ASME/ANS-RA-Sb-2013
Other Hazards PRA (X)	Similar to X in ASME/ANS-RA-Sb-2013
Event Sequence Quantification (ESQ)	Similar to QU in ASME/ANS-RA-Sb-2013 except that the event sequences are mapped to user defined end states and release categories and cover anticipated events, and events within and beyond the design basis, and accidents involving single reactor units and multiple reactor units
Mechanistic Source Term Analysis (MS)	Similar to source term requirements in ANS Level 2 PRA standard[45] except that source terms cover both single unit and multiple reactor units
Radiological Consequence Analysis (RC)	Similar to the requirements in the ANS Level 3 PRA standard[75] except that there is an option to limit the scope to the performance of site boundary dose calculations rather than a full Level 3 analysis
Risk Integration (RI)	This PRA element is unique to the non-LWR PRA standard and includes requirements to combine the results of the ESQ and RC elements to affect an integrated risk assessment with options to combine the information in different ways. This includes requirements to establish the risk significant release categories which is then used in ESQ to decompose the risk significant accident sequences and basic events.

The key differences between the non-LWR and the supporting LWR standards are listed below:

- When the non-LWR PRA is developed to its fullest scope, the event sequences are developed sufficiently to establish a comprehensive set of release categories each with a mechanistic source term, a quantification of radiological consequences, frequencies, and risk.
- Core damage frequency is not used because existing definitions of CDF are in terms of LWR characteristics (liquid level in the reactor vessel, oxidation temperature of zircalloy cladding, metallic fuel melting, etc.) that may not have a counterpart in the non-LWR plant. As explained more fully in Section 3.7, this standard includes technology neutral risk metrics including frequency and site boundary dose consequences for each accident family or LBE, the individual risk metrics for the NRC safety goal QHO comparisons, and complementary cumulative distribution curves (frequency of exceeding selected consequence metrics). The standard includes provisions and requirements for user defined reactor-specific metrics.
- The non-LWR standard includes requirements to support a multi-unit PRA for multi-unit and modular reactor designs. These requirements are included to enable the PRA to provide risk insights to design features to effectively manage the risks of accidents involving multiple reactor modules.
- PRA peer reviews for advanced non-LWRs are expected to confirm that technical requirements used in the PRA are met as will the current LWR PRA peer reviews. An emphasis of these reviews will focus on elements that are different than LWR PRAs arising from fundamental differences in the safety design approach as well as limitations of PRAs that are performed during preoperational phases. The major elements that are different include the event sequence end states and risk metrics, approach to evaluating success criteria, treatment of technology specific phenomena, and treatment of uncertainty due to lack of experience with the reactor technology. Consideration should be given to “in process” reviews rather than reviews applied near the completion of the PRA.

4.2.2 Additional Guidance for PRA Technical Adequacy

Useful guidance in developing PRA models for Advanced Non-LWRs is found in NUREG-1860.[31] This document includes a good summary of the safety characteristics of non-LWRs, guidance on how to use the PRA to inform the selection of LBEs and SSC special treatment requirements, and high level requirements for PRA technical adequacy that parallel and supplement the requirements in the ASME/ANS RA-S-1.4-2013.[43] The Writing Group responsible for the non-LWR standard plans to review the PRA material in NUREG-1860 for consideration as input in the revised non-LWR PRA standard.

The IAEA has published attributes of a full scope Level 1 PRA in Reference [48]. Although this reference was developed for currently operating LWR plants, it includes attributes for supporting a multi-reactor PSA which are similar to those in ASME/ANS RA-S-1.4-2013. Additional references to support PRA modelling of multi-reactor module plants are listed in Section 4.1.1.

PRAs are required to support Generic Design Reviews for new facilities and periodic safety reviews for operating plants licensed in the United Kingdom. These reviews are similar to Design Certifications Applications reviews performed by the NRC. The Safety Assessment Principles (SAPs) used by the United Kingdom licensing authorities include specific requirements for the required PRAs and also serve as useful guidance for technical adequacy of a PRA.[54] Importantly, the SAPs are reactor technology neutral because they support the licensing of currently operating gas-cooled reactors and LWRs and future licensing of advanced non-LWRs.

4.3 Insights from B. John Garrick Institute for the Risk Sciences

The LMP team includes the B. John Garrick Institute for the Risk Sciences at the University of California, Los Angeles. To support the development of the LMP white papers on RIPB decision making, the Garrick Institute was tasked with reviewing the NGNP white papers on PRA, LBE selection, SSC safety classification, and DID. The Garrick Institute review scope included the review of earlier drafts of this LMP paper on PRA development as well as the LMP white paper on LBE selection.[2] Insights from these reviews have been incorporated into this paper and include the following key points:

- The Garrick review was generally supportive of the approaches reflected in the LMP and PRA white papers and recognize advancements beyond the NGNP papers.
- It was noted in this review that the examples provided for PRA development and LBE selection are from modular HTGRs including pebble bed reactors and sodium cooled fast reactors, which benefit from a long and rich history of design and PRA development. Additional guidance is needed for other reactor concepts such as molten salt reactors which are at an earlier stage of design development and lack a significant body of PRA case studies. An effort was made in preparation of this paper to provide more guidance on how to begin the PRA development, which is found in Section 3 of this paper.
- It was noted that some molten salt reactor concepts have radioactive waste processing systems with much greater radionuclide inventories including tritium than those for operating LWR plants. Releases from these systems may pose greater risks than those from reactor based source terms. In addition, some of the coolants used in these reactors may pose non-radiological and toxic hazards which need to be addressed.
- Use of plant-year vs. reactor-year based frequency metrics and the explicit inclusion of multiple reactor and multi-source accident sequences are recognized as a strength of the LMP approach.
- Lessons from the Fukushima Daiichi accident need to be addressed including the important role of recovery actions and realistic treatment of accident management. It is not clear whether mitigation strategies (such as the “FLEX” capabilities developed by the U.S. power industry in response to the accident) will be used and how these will be addressed in the PRA.
- Risk metrics may need to be modified for plants located near industrial facilities providing process heat. In addition, there are risk metrics beyond those used to select and evaluate

LBEs that greatly expand the capabilities of PRA for risk management. Examples are facility investment risk, plant production risks, and land contamination based risk metrics.

- Additional guidance on the treatment of uncertainty in estimating frequencies and consequences for new reactors would be beneficial.
- More guidance on the necessary tools for thermo-fluid and neutronic response of plants to event sequences as well as the prediction of mechanistic source terms would be beneficial. The LMP plans to provide a separate white paper on analytical tools to address this comment.
- John Garrick, recognized as one of the pioneers of PRA technology, has provided comments that are provided as part of the Garrick Institute review. He offers counsel not to use the PRA standards in prescriptive manner which may inhibit the creative work needed to fully develop the PRA technology for new reactors. He reminds us that the risk triplet,[*] defined in his landmark paper with Stan Kaplan, On the Quantitative Definition of Risk,[74] has been successfully applied to many different reactor and non-reactor technologies unencumbered by prescriptive standards. Finally, he recommends that treatment of uncertainty and external events be introduced at an early stage and objects to the artificial separation of PRA models by hazard group. Rather he advocates the concept of a fully integrated risk assessment as having the best opportunity to yielding the most effective risk management strategies early in the design process.

4.4 NRC Roles in Ensuring PRA Technical Adequacy

Successful RIPB licensing decisions within the LMP framework will require active participation by the NRC. This participation is recommended to include the following:

- Continued NRC staff participation on the ASME/ANS JCNRM Writing Group responsible to the Advanced non-LWR PRA standard
- Review and endorsement of the ANSI version of the non-LWR PRA standard scheduled for balloting in 2018
- Participation on PRA peer review teams required by the PRA standards

Review of licensing documents in which information from the PRA is used to support RIPB decisions such as selection of LBEs, safety classification and special treatment requirements of SSCs, principal design criteria informed by the PRA, and use of the PRA to affect a risk-informed evaluation of DID.

[*] The risk triplet, according to the definition of Kaplan and Garrick, is comprised of a structured set of scenarios, quantitative estimates of the frequency and consequences of each scenario, and a probabilistic representation of the uncertainty in these estimates. The risk assessment results from answering three fundamental safety questions: “What can go wrong?”, “How likely is it?”, and “What are the Consequences?”.

5.0 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 the use of PRA in the selection and evaluation of LBEs.

The LMP objective is to assist the NRC to develop regulatory guidance for licensing advanced non-LWR plants. In this paper, the LMP is seeking:

1. NRC's approval of the proposed technology-inclusive PRA approach for incorporation into appropriate regulatory guidance for advanced non-LWRs
2. Identification of any issues that have the potential to significantly impact the use of risk insights derived from the PRA in the selection and evaluation of LBEs and safety classification of SSCs

Outcome objectives for additional uses of the PRA in RIPB decisions beyond LBE selection will be addressed in a future LMP paper.

The LMP is seeking agreement on the following specific statements regarding the PRA approach:

1. The scope and technical approach for advanced non-LWR PRAs outlined in this paper are appropriate for the intended applications of the PRA in the construction and operating license application for advanced non-LWR plants including modular HTGRs, molten salt reactors, sodium reactors, and other advanced non-LWR concepts. These PRA applications include input to:

   - Evaluation of design alternatives and incorporation of risk insights into the design
   - Selection of LBEs including the DBAs
   - Safety classification and special treatment requirements of SSCs
   - Selection of performance targets for the reliability and capability of SSCs within the scope of the PRA
   - RIPB evaluation of DID adequacy

   LMP Approach
   A technology-inclusive approach to performing a PRA on a non-LWR plant is described in this paper and has been demonstrated using examples for an HTGR and SFR design.

2. The road-map presented in this paper for introducing the PRA at an early stage in the design and progressively increasing the scope and level of detail of the PRA models and documentation consistent with the scope and level of detail of the supporting design and siting characterization is appropriate. The iterative nature of the PRA and design

development creates a need to review and revise the supported RIPB decisions to incorporate new risk insights.

LMP Approach

The LMP approach to developing the PRA is to introduce the PRA at an early stage of design to ensure that the safety design approach benefits from risk insights at time when the design can be effectively optimized. The PRA is initially simplified, high level, and limited in scope and then is periodically upgraded and expanded in scope and detail as the design matures and site information becomes available. This leads to an iterative design, PRA, and LBE definition process.

3. The TI approaches to IE selection, event sequence development, end-state definition, definition of risk metrics, definition of risk importance measures, and approach to risk-significance determination outlined in this paper are technically adequate for the intended PRA applications.

LMP Approach

The TI approach that has been developed for PRA modeling and quantification is appropriate for the full spectrum of advanced reactor concepts currently being considered for deployment. The risk metrics selected are capable of supporting the intended applications of the PRA as demonstrated for two example non-LWR plants, an HTGR and an SFR.

4. The TI approaches to the treatment of inherent characteristics and passive SSCs outlined in this paper are technically adequate.

LMP Approach

The TI approach for the treatment of inherent and passive safety features that is proposed in this paper is consistent with the approach used in PRAs for evolutionary LWR plants.

5. The TI approach to using deterministic engineering analyses for assessing the plant response to IEs and event sequences, success criteria, and mechanistic source terms is appropriate for the proposed risk-informed, performance-based advanced non-LWR design and licensing approach.

LMP Approach

The LMP approach for interfacing the PRA development with supporting deterministic engineering analyses is consistent with PRAs on operating and evolutionary LWRs, but require reactor and design specific analytical tools for performing plant response modeling, success criteria development, and mechanistic source term development.

6. The TI approach to the development of PRA data outlined in this paper, including the use of applicable data from non-nuclear sources, LWRs, expert opinion, and treatment of uncertainty, is a technically adequate approach for the advanced non-LWR PRA.

LMP Approach

The LMP approach for PRA database development has been described in this paper. The challenges to developing this data will differ among the various non-LWR types, however these challenges are capable of being met making use of applicable LWR and non-LWR data sources, generic non-nuclear sources, expert opinion, and adequate treatment of uncertainties.

7. The TI process for PRA treatment of uncertainties in the estimation of accident frequencies and the quantification of mechanistic source terms and consequences in the PRA is a technically adequate approach for the purpose of developing and analyzing the results of the PRA.

LMP Approach

Most of the available guidance on PRA treatment of uncertainties is technology-inclusive. The supporting PRA standards emphasize the need to identify and evaluate sources of uncertainty in PRA model inputs and in quantifying the selected risk metrics. The available case studies on non-LWR PRAs described in this paper have demonstrated that this capability exists.

8. The TI approach for the PRA treatment of multi-unit or multi-module plants including the delineation of accidents involving single and multiple reactor modules and radiological sources is technically adequate to support licensing of single and multi-module plant configurations. It is recognized that case studies in the application of PRA to non-core source of radioactive material are lacking.

LMP Approach

In the PRA developments following the Fukushima Daiichi accident, it is extremely important that its lessons are taken into account. Effective risk management strategies to control the risks of multi-reactor and multi-source accidents require that the associated risks be sufficiently characterized. The LMP approach to treatment of multi-reactor and multi-source accidents benefits from experience in addressing this issue in previous LWR, SMR, and non-LWR PRAs including those on the MHTGR and PRISM.

9. The approach to establishing the technical adequacy of the PRA for its intended RIPB applications based on the ASME/ANS PRA Standard for Advanced non-LWRs[43] and supporting peer reviews as described in this standard is acceptable. It is recommended that NRC take an active role in contributing to, reviewing, and endorsing this standard when the trial use period is completed and the ANSI version of this standard is developed.

LMP Approach

The LMP approach to addressing technical adequacy is based on using the ASME/ANS non-LWR PRA Standard to develop the base PRA and support the necessary peer reviews, and then to justify RIPB licensing decisions applications as part of the license application and supporting NRC staff licensing reviews.

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