Use of PRA to Select Licensing Basis Events

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Abstract: The purpose of this paper is to summarize key elements of the risk-informed and performance-based methods developed within the Industry led Licensing Modernization Project (LMP). The LMP is jointly sponsored by the U.S. Department of Energy and the U.S. nuclear industry to assist the U.S. Nuclear Regulatory Commission (NRC) in the development of regulatory guidance for advanced non-light water reactors currently under development in the U.S. The purpose of this paper is to summarize a risk-informed and performance based approach for the selection and evaluation of LBEs for advanced non-LWRs. This paper summarizes the approach which builds on a PRA model that is introduced early in the design process and provides examples for selected advanced non-LWR technologies.
Keywords: PRA, Licensing Basis Events (LBEs), risk-informed and performance-based (RIPB).

1. INTRODUCTION

Many of the current regulatory requirements for US nuclear power plants are based on light water reactor (LWR) technology. To facilitate efficient, effective, and predictable licensing expectations for a spectrum of novel, advanced, non-LWRs additional regulatory guidance is needed. 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 implementing the proposals. These proposals are described in a series of papers that have been submitted for review by the NRC [1][2][3][4] . The LMP approaches for defining, selecting, and evaluating LBEs build on those that were proposed for the Next Generation Nuclear Plant (NGNP [5]. These papers are currently being used to assist the NRC in development of regulatory guidance for licensing advanced non-LWR plants.

The technology-inclusive, risk-informed, and performance-based (TI-RIPB) approach to selecting LBEs is designed to ensure that an appropriate set of events are identified for each reactor design and technology to support design and licensing decisions. The objective of this approach is the identification of the most risk significant events for each design and technology to ensure that the appropriate limiting events can be selected for establishing the design and licensing bases. This is essential to ensure that risk insights are appropriately reflected in the design and licensing decisions. These decisions include not only the selection of the DBAs, but also the events to be considered in the formulation of design criteria for structures, systems, and components, and licensing requirements for protection against external hazards, siting evaluations, and emergency planning.

In the development of a suitable approach for selection and evaluation of LBEs a set of desirable attributes was adopted. According to these attributes, the process for selecting LBEs for advanced non-LWRs should be:

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

2. OVERVIEW OF LBE SELECTION AND EVALUATION PROCESS

2.1. Definition of LBEs

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

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

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

2.2. LBE Selection and Evaluation Flowchart

A flow chart indicating the steps to identify and evaluate LBEs in concert with the design evolution is shown in Figure 1. These steps are intended to be carried out by the design and design evaluation teams responsible for establishing the key elements of the safety case and preparing a license application. The process is used to prepare an appropriate licensing document, e.g., licensing topical report, that documents the derivation of the LBEs, which would be reviewed by the regulator as part of license review. The design and design evaluation teams are responsible for selecting the LBEs and justifying their selections. The regulator is responsible to review the design, the LBE selections, and their derivation. The actual selection of LBEs is performed in Tasks 1-6, while the evaluations of the LBEs which are comprised of both probabilistic and deterministic elements are performed in Task 7. Tasks 8,9, and 10 reflect the fact that LBE selection and evaluation are an iterative process and are also repeated at different stages of design, siting, and licensing.

The process is implemented in the following LBE selection tasks:

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

Task 2 Design Development and Analysis

The design development is performed in phases and often includes pre-conceptual, conceptual, preliminary, and final design phases and may include iterations within phases. The design development and analysis includes definition of the key elements of the safety design approach, the design approach to meet the top level design requirements for energy production and investment protection, and analyses to develop sufficient understanding to perform a PRA and the deterministic safety analyses. The subsequent Tasks 3 through 10 are repeated for each design phase until the list of LBEs is finalized.

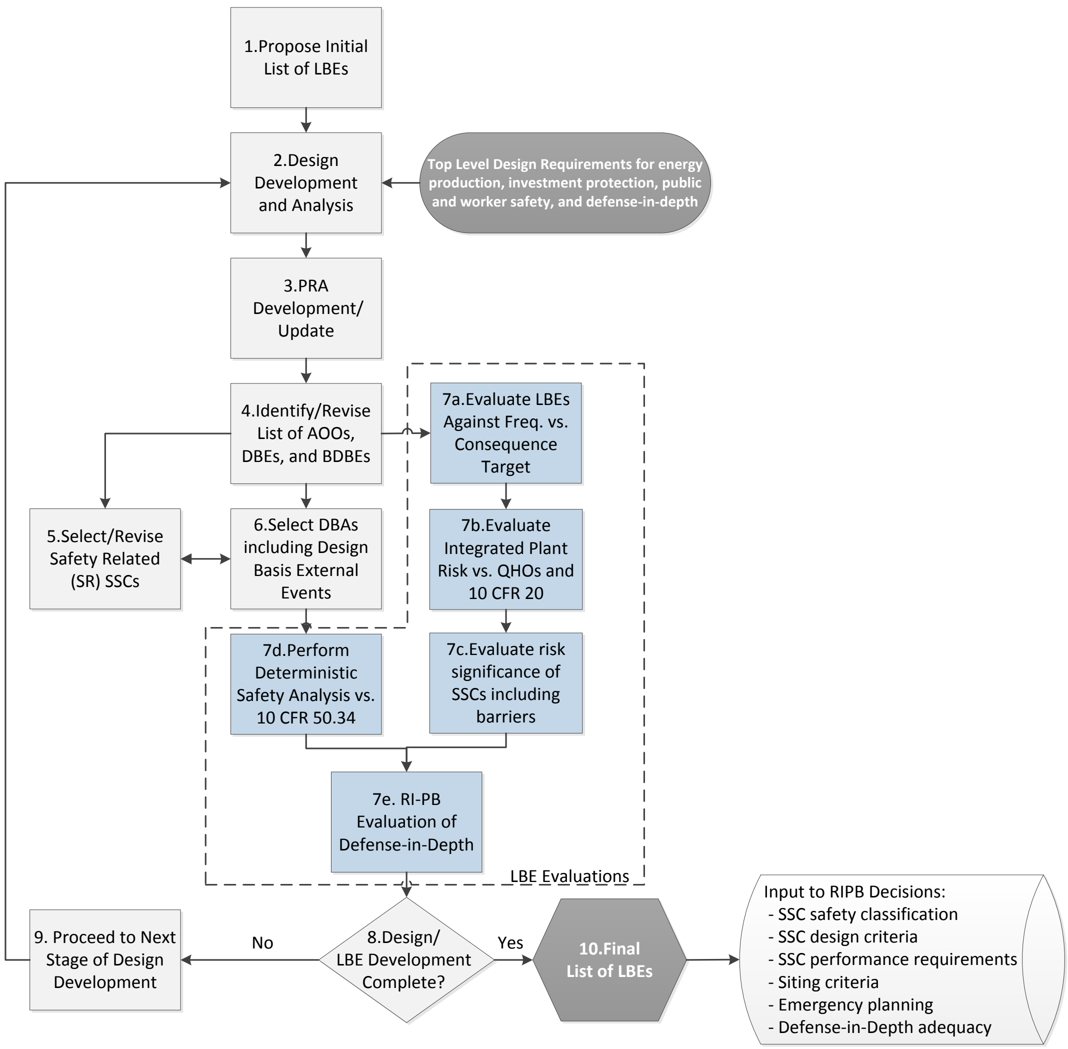


Figure 1: LMP Process for Selecting and Evaluating Licensing Basis Events

Task 3 PRA Development/Update

A PRA model is developed and updated for each phase of the design. 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 failure modes and effects analysis (FMEA) and process hazard analysis (PHA), provide industry-standardized and established practices to ensure that early stage evaluations are systematic, reproducible and as complete as the current stage of design permits. In the first design phase, which is typically the pre-conceptual design, the PRA is of limited scope, uses simplified models and coarse level of detail and makes use of engineering judgment to fill in missing details much more than a completed PRA that would be of sufficient scope and level of detail to meet applicable PRA standards. The scope and level of detail of the PRA are then enhanced as the design matures and siting information is defined. More details on the role of PRA in the selection and evaluation of LBEs are provided in Section 3 of this paper.

Task 4 Identify/Revise List of AOOs, DBEs, and BDBEs
The event sequences modeled and evaluated in the PRA are grouped into accident families each having a similar initiating event, challenge to the plant safety functions, plant response, and mechanistic source term if there is a release. Many LBEs do not involve a release but are needed to identify challenges to the plant safety functions and the capabilities of SSCs needed to prevent and mitigate accidents. AOOs are off-normal events that are expected to occur in the life of the plant with frequencies exceeding 10-2 per plant-year, where a plant may be comprised of multiple reactor modules. DBEs are less frequent events that may be expected to occur in the lifetime of a fleet of plants with frequencies between 10-4 to 10-2, per plant-year. BDBEs are rare events with frequencies less than 10-4/plant-year but with upper bound frequencies greater than 5x10-7/plant-year. LBEs may or may not involve release of radioactive material and may involve two or more reactor modules or radionuclide sources. LBEs with no release are important to identify challenges to SSCs, including barriers, that are responsible for preventing or mitigating a release of radioactive material.

Tasks 5 and 6 are performed together rather than sequentially. Examples are provided in this paper for the MHTGR and PRISM to illustrate how this tasks were implemented for these reactors.

Task 5 Select Safety-Related SSCs
In Task 5 the full set of DBEs are examined to identify the safety functions that are necessary and sufficient to meet the F-C target for all DBEs and to conservatively ensure that 10 CFR 50.34 dose requirements can be met. For each of these required safety functions, the design team makes a decision on which SSCs that perform these required safety functions for should be classified as safety related for the full set of DBEs.

Task 6 Select DBAs
For each DBE identified in Task 4, a DBA is defined that includes the required safety function challenges represented in the DBE, but assumes that the required safety functions are performed exclusively by safety-related SSCs selected in Task 5. These DBAs are then used in Chapter 15 of the license application for supporting the deterministic safety analysis using conservative or best estimate approaches with appropriate accounting for uncertainties.

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

Task 7a. Evaluate LBEs against Frequency – Consequence (F-C) Target
In this task the results of the PRA which have been organized into LBEs will be assessed using the F-C Target shown in Figure 2. The figure does not define specific acceptance criteria for the analysis of LBEs but rather provides a tool to focus the attention of the designer and those reviewing the design and related operational programs to the most significant events and possible means to address those events. The NRC's Advanced Reactor Policy Statement includes expectations that advanced reactors will provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions. The safety margin between the design-specific PRA results, and the F-C Target provides one useful and practical demonstration of how the design fulfills the Commission expectations for enhanced safety. These margins also are useful in the evaluation of defense-in-depth adequacy in Step 7e. The evaluations in this step are performed on each LBE separately. The mean values and the uncertainties in the frequency and consequence estimates are used to classify the LBEs into AOOs, DBEs, and BDBE categories. Part of the LBE frequency-dose evaluation is to ensure that LBEs involving releases from two or more reactor modules do not make a significant contribution to risk and to ensure that measures to manage the risks of multi-module accidents are taken to keep multi-module

releases out of the list of DBAs. Another key element of this step is to identify design features that are responsible for limiting the frequency-dose results, including those that are responsible for preventing any release for those LBEs where applicable. This evaluation leads to performance requirements and design criteria that are developed within the framework of the SSC classification step in the TI-RIPB design and licensing approach.

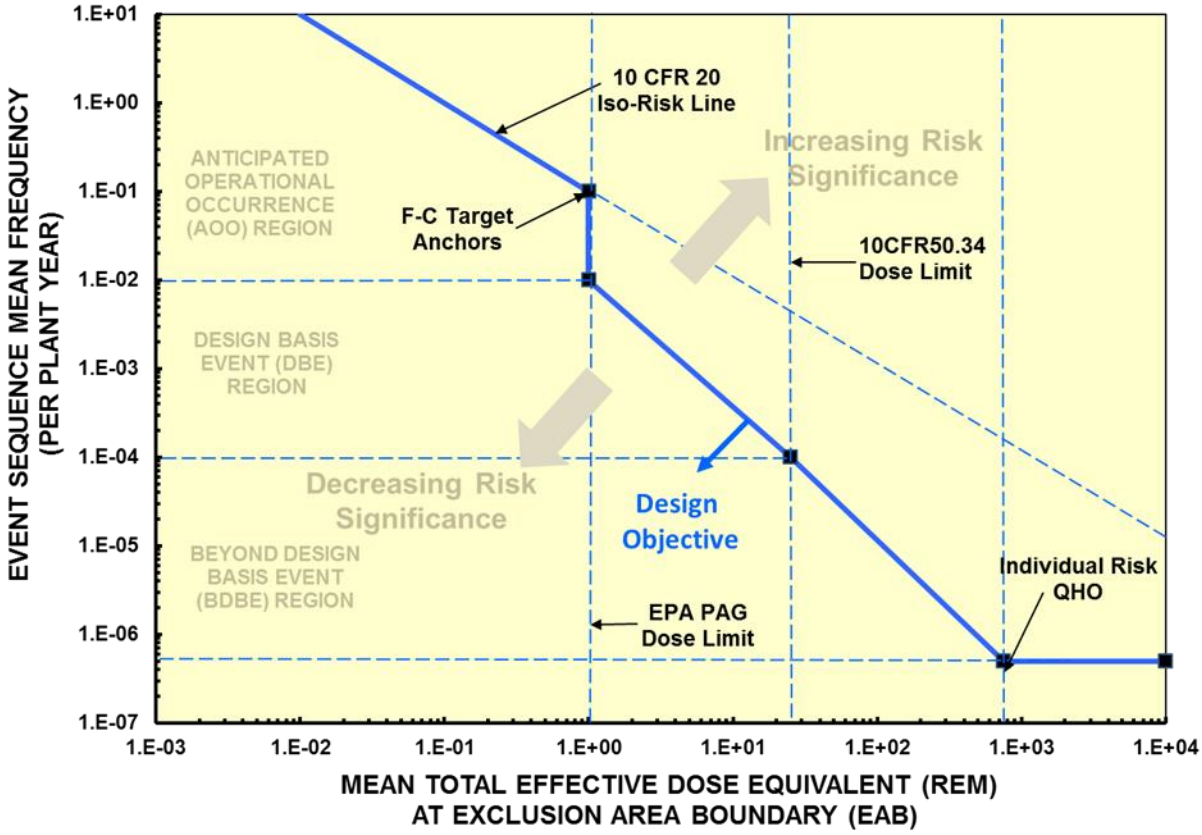


Figure 2: Frequency-Consequence Target Proposed for Evaluating Risk Significance of LBEs

Task 7b. Evaluate Integrated Plant Risk

In this task, the integrated risk of the entire plant, which may consist of multiple reactor modules, is evaluated against three cumulative risk targets as follows:

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

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

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

Task 7d. Perform Deterministic Safety Analyses against 10 CFR 50.34
This task corresponds to the traditional deterministic safety analysis that is found in Chapter 15 of the license application. Deterministic analyses often use stylized scenarios for the purpose of demonstrating compliance with specific requirements, establish safety margins, and define equipment specifications and operational limits. Deterministic safety analysis can use conservative or best-estimate analytical methods with an appropriate accounting of uncertainties. The uncertainty analyses in the mechanistic source terms and radiological doses that are part of the PRA are available to inform the conservative assumptions used in this analysis and to avoid the arbitrary "stacking" of conservative assumptions that lack physical meaning.

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

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

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

Task 10 Finalize List of LBEs and Safety Related SSCs
Establishing the final list of LBEs and safety related SSCs signifies the completion of the LBE selection process and the selection of the safety related SSCs. The next step in implementing the TI-RIPB approach is to formulate performance requirements and regulatory design criteria for SSCs that are necessary to limit LBE frequencies and doses. Important information from Task 7b is used for this purpose. The results of the event selection process and related evaluations will be described in appropriate sections of the Safety Analysis Report.

3. ROLE OF PRA IN THE SELECTION AND EVALUATION OF LBES

3.1. Summary of PRA Roles

The LMP framework for selecting and evaluating LBEs and supporting the development licensing requirements is a risk-informed and not a risk-based process. The LBE process outlined in Figure 1 is clearly risk-informed because it relies on traditional deterministic elements as well as deriving important inputs from a reactor technology and design specific PRA. The PRA itself is developed on a foundation of traditional deterministic elements so even the PRA model is risk informed as will be explained more fully in this section of the paper.

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 the F-C Target in Figure 2.
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. Determining the risk significance and 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 cumulative risk targets.
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

It is noted that the development of the PRA is not developed in a single step, but rather in an iterative process that proceeds with the successive stages in the development of the design. Each of the decisions associated with the above applications is reviewed as the PRA is updated and upgraded and revised as needed to incorporate changes to the state of knowledge about risks.

3.2. Roadmap for PRA Development as Design Matures

The LMP framework recommends that PRA be introduced at an early stage of design. To optimize the benefits and to obtain the best return on the PRA investment, it is recommended that the development of the PRA be introduced before the completion of the conceptual design.

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 as needed to meet current PRA requirements for new reactors 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 and to begin the iterative process of ensuring that risk targets have been achieved with sufficient margins. When available to support pre-licensing interactions, an understanding of how risk insights have been incorporated into the design will help set the foundation for subsequent risk-informed licensing decisions.

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 [6]. 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 recognized as an integral part of the PRA methodology. Basic steps in a HAZOPs evaluation are shown in Figure 3 adapted from Reference [6].

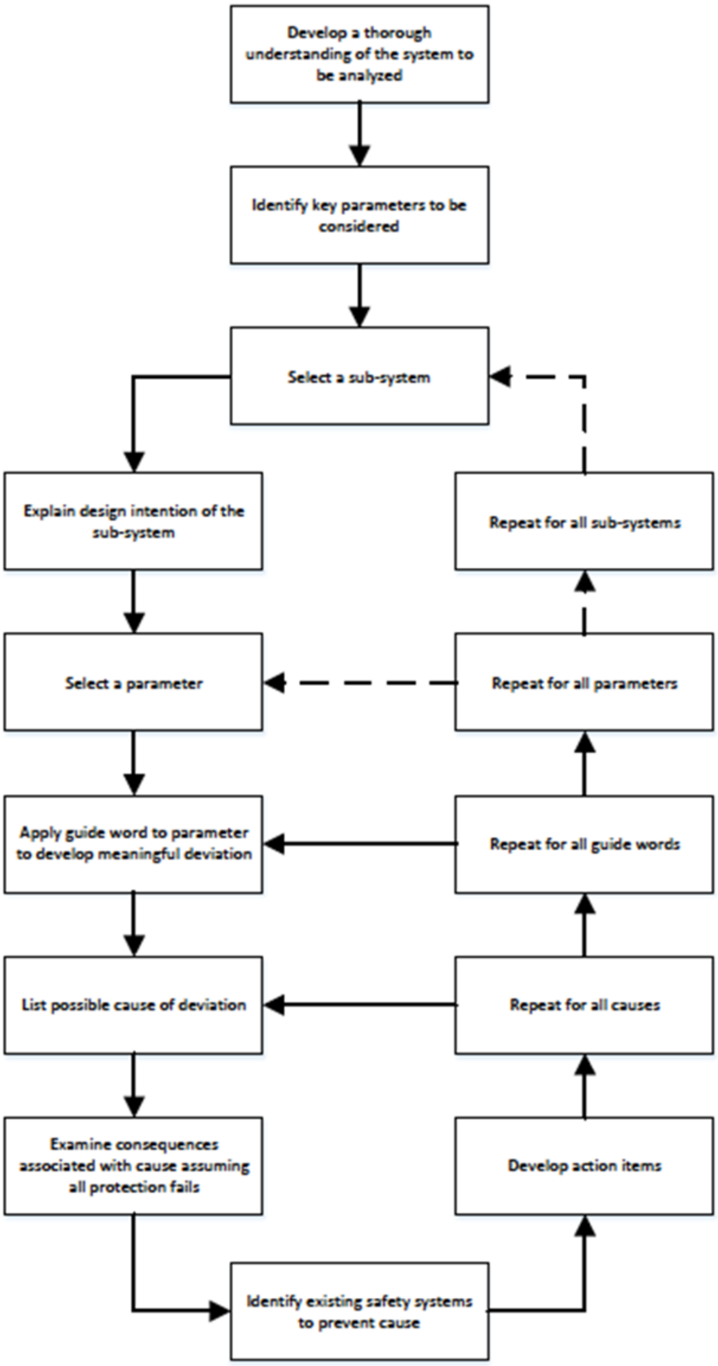


Figure 3: Basic Steps in HAZOP Evaluation

When early stage safety analyses such as HAZOP and FMEA are introduced in the pre-conceptual or early stages of the conceptual design, and the initial PRA development begun during this timeframe, the PRA model 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

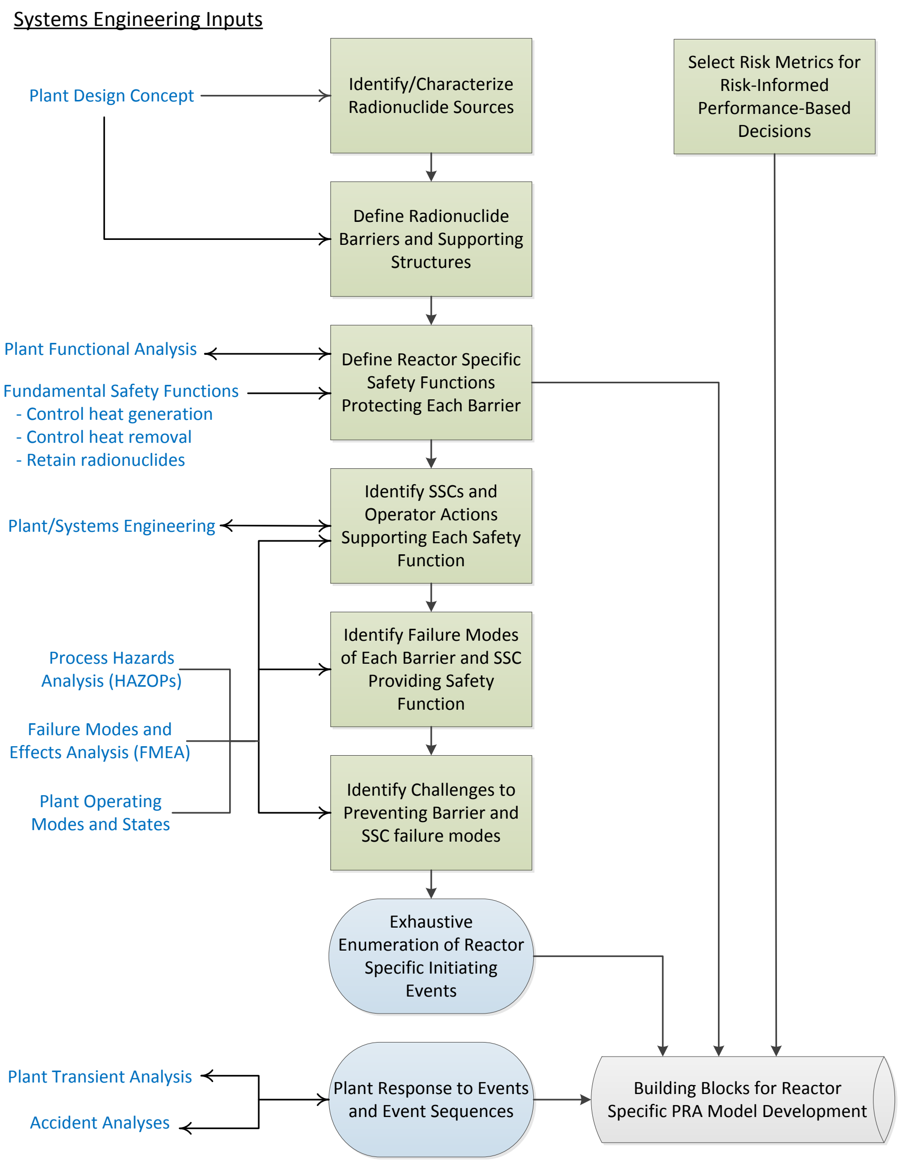
Despite these simplifications, the PRA development would be capable of defining a reasonably complete set of event sequences and order of magnitude estimates of the frequencies and site boundary doses of those involving a release. Hence the PRA should be sufficient to develop an initial set of LBEs to 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 initial PRA model is developed in close coordination with the design development and gets input from a number of design analyses and early safety analysis methods that comprise deterministic inputs to the PRA. Several of these key PRA-Design-Analysis interfaces are illustrated in Figure 4. When the deterministic inputs are modified revised as the design evolves, the PRA models are modified as appropriate.

3.3. Absolute Risk Metrics

The traditional approach to evaluating risk significance, only the relative importance of each accident sequence or basic event normalized against a baseline total risk is considered. 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 corresponding estimates 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 of accident sequences and basic events can be compared against the risk goals or targets rather than the baseline risks. One example of the use of absolute risk metrics is the approach to defining risk significance LBEs using the F-C Target in Figure 1. Another example is that identified in the LMP SSC paper for establishing the risk significance of SSCs [3]. For this metric, risk significant SSCs if any of the following criteria are met:

- A prevention or mitigation function of the SSC is necessary to meet the design objective of keeping all LBEs within the F-C target. This is determined by assuming failure of the SSC in performing and prevention or mitigation function and checking on how the resulting LBE risks compare with the F-C target. The LBE is considered within the F-C target when a point defined by the upper 95%-tile uncertainty of the LBE frequency and dose estimates are within the F-C target.
- The SSC makes a significant contribution to one of the cumulative risk metrics used for evaluating the risk significance of LBEs. A significant contribution to each cumulative risk metric limit is satisfied when total frequency of all LBEs with failure of the SSC exceeds 1% of the cumulative risk metric limit . The cumulative risk metrics and limits include:
  - The total frequency of exceeding of a site boundary dose of 100 mrem < 1/plant-year (10 CFR 20)

- The average individual risk of early fatality within 1 mile of the Exclusion Area Boundary (EAB) < 5×10^{-7}/ plant-year (QHO)
- The average individual risk of latent cancer fatalities within 10 miles of the EAB shall not exceed 2×10^{-6}/plant-year (QHO)

3.4. PRA Technical Adequacy

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

The approach to addressing each of these challenges is addressed in the LMP PRA paper [1]. The PRA models for multi-module designs will be addressed in manner that explicitly includes event sequences involving both single and multiple reactor modules and radionuclide sources. The development of a database for estimating initiating event frequencies, component failure rates, common cause modeling parameters, maintenance unavailability, etc. will benefit from the availability of relevant and applicable data from existing plants and other industries and data already developed for advanced non-LWR designs. Treatment of uncertainty will need to be emphasized in view of the lack of relevant service experience, but this challenge is not unlike the challenge faced by Rasmussen in the development of the first LWR PRA which did not use any data from operating plants. The treatment of inherent and passive safety features will benefit from PRAs that have already been development for advanced non-LWRs and SMRs that have reduced the dependence on active systems. The new risk-informed applications present their own challenges that experience is expected to resolve.

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 [7]. 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.

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 [8].
- A PRA 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.[9]
- 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.

4. CONCLUSION

The LMP has developed a systematic and reproducible process for defining, selecting, and evaluating LBEs for advanced non-LWRs. The approach makes use of traditional engineering approaches as well as a technology and design specific PRA to risk-inform the selections and evaluations. The PRA models intended for this application have additional roles in the safety and risk significance classification of SSCs and in the risk-informed and performance based evaluation of defense-in-depth adequacy.

Acknowledgements

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References

[1] Idaho National Laboratory, "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, Selection of Licensing Basis Events," Draft, April 2017.[Adams Accession Number ML17104A254]

[2] Idaho National Laboratory, "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, Probabilistic Risk Assessment Approach," Draft, June 2017.[Adams Accession Number ML17158B543]

[3] Idaho National Laboratory, "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, Safety Classification and Performance Criteria for Structures, Systems and Components," Draft, October 2017.[Adams accession number ML17290A463]

[4] Idaho National Laboratory, "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, Risk-Informed and Performance-Based Evaluation of Defense-in-Depth Adequacy," Draft, December 2017.[Adams accession number ML17290A463]

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

[6] American Institute of Chemical Engineers, "Guidelines for Hazard Evaluation Procedures," by the Center for Chemical Process Safety, the (2008).

[7] American Society of Mechanical Engineers and American Nuclear Society, "Probabilistic Risk Assessment Standard for Advanced non-LWR Nuclear Power Plants," RA-S-1.4-2013.

[8] GE Hitachi Nuclear Energy, "Final Scientific/Technical Report: Development/Modernization of an Advanced Non-LWR Probabilistic Risk Assessment," Federal Grant DE-NE0008325, 2017.

[9] Zhang, S. et al., An Integrated Modeling Approach for Event Sequence Development in Multi-Unit Probabilistic Risk Assessment," Reliability Engineering and System Safety, Volume 155, November 2016, pp. 147-159.