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# NUCLEAR TECHNOLOGIES AND MATERIALS ADVANCED REACTOR CONCEPTS-20

# FAST MODULAR REACTOR SELECTION OF LICENSING BASIS EVENTS

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GENERAL ATOMICS
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## ACRONYMS

| Acronym | Definition |
|---|---|
| ALARA | As Low As Reasonably Achievable |
| ANS | American Nuclear Society |
| AOO | Anticipated Operational Occurrence |
| ARC-20 | Advanced Reactor Concepts-20 |
| ARDP | Advanced Reactor Demonstration Program |
| ASME | American Society of Mechanical Engineers |
| BDBE | Beyond Design Basis Event |
| COL | Combined License |
| DBA | Design Basis Accident |
| DBE | Design Basis Event |
| DID | Defense-In-Depth |
| DOE | Department of Energy |
| EAB | Exclusion Area Boundary |
| EDG | Emergency Diesel Generator |
| EPA | Environmental Protection Agency |
| EPRI | Electric Power Research Institute |
| EPZ | Emergency Planning Zone |
| ESP | Early Site Permit |
| F-C | Frequency-Consequence |
| FMR | Fast Modular Reactor |
| GA-EMS | General Atomics Electromagnetic Systems |
| GFR | Gas-cooled Fast Reactor |
| GT-MHR | Gas Turbine-Modular Helium Reactor |
| HPME | High Pressure Melt Ejection |
| IAEA | International Atomic Energy Agency |
| IE | Initiating Event |
| LBE | Licensing Basis Event |
| LMP | Licensing Modernization Project |
| LPZ | Low Population Zone |
| LWR | Light Water Reactor |
| MHTGR | Modular High-Temperature Gas-Cooled Reactor |
| MWe | Megawatt electric |
| non-LWR | non-Light Water Reactor |
| NRC | Nuclear Regulatory Commission |
| PAG | Protective Action Guide |
| PCS | Power Conversion System |
| PCU | Power Conversion Unit |
| PRA | Probabilistic Risk Assessment |

| Acronym | Definition |
|---|---|
| QHO | Quantitative Health Objective |
| RCPB | Reactor Coolant Pressure Boundary |
| RHR | Residual Heat Removal |
| RVCS | Reactor Vessel Cooling System |
| RG | Regulatory Guide |
| SRM | Staff Requirements Memorandum |
| SSC | Structure, System, and Component |
| TCG | Turbine-Compressor-Generator |
| TEDE | Total Effective Dose Equivalent |
| TI-RIPB | Technology Inclusive – Risk-Informed, Performance-Based |
| TLRC | Top-Level Regulatory Criteria |



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## 1. INTRODUCTION

General Atomics Electromagnetic Systems (GA-EMS) is developing a helium-cooled Fast Modular Reactor (FMR) [Reference 1]. The project has been selected by the U.S. Department of Energy (DOE) for Advanced Reactor Concepts-20 (ARC-20) under Advanced Reactor Demonstration Program (ARDP). The long-term goal is to design, license, and commercialize the FMR plant by the mid-2030s. To achieve the goal of licensing the FMR, GA-EMS has been engaged with the Nuclear Regulatory Commission (NRC) from the initial stage of the project.

A fundamental aspect of the licensing process is the development of a comprehensive licensing basis. This entails creating a collection of documents and technical criteria that will serve as the foundation upon which the NRC will grant a license for the construction and operation of the nuclear facility. The NRC requires reactor designs to be evaluated based on several different kinds of events that are considered part of the licensing basis. Licensing basis events (LBEs) are certain event sequences that are chosen to be considered in the design of a nuclear power plant. As an effort to support the FMR pre-application regulatory engagement plan, GA-EMS is developing a process of LBE selection applicable to the FMR design.

The licensing basis essentially outlines the framework for evaluating the safety and feasibility of a plant. LBEs can be understood as key event sequences that are essential for establishing the safe operating parameters and equipment safety classification. These events typically encompass a range of scenarios, such as loss-of-coolant accidents, loss-of-power events, equipment failures, or other potential hazards that could potentially impact the safe operation of the plant.  LBEs refer to the events or accidents that a nuclear power plant's design and operation must be able to safely manage and mitigate to prevent harm to the public and the environment. LBEs may affect the safety or environmental protection measures required by the licensing basis for a nuclear plant.

LBEs are categorized as follows:

- Anticipated Operational Occurrences (AOOs)
- Design Basis Events (DBEs)
- Design Basis Accidents (DBAs)
- Beyond Design Basis Events (BDBEs)

The process for selecting LBEs refers to the procedure for identifying the types of events that a nuclear plant should be designed and operated to withstand. Selection of LBEs for the FMR design will follow guidance provided in NEI 18-04 [Reference 2]. This guidance provides an integrated and highly interdependent methodology for identifying and evaluating licensing basis events, classifying, and establishing performance criteria for SSCs, and evaluating defense-in-depth (DID) for advanced reactor designs. The process of LBE selection following this guidance will demonstrate the effectiveness of the selection process to facilitate the determination of risk significant LBEs and SSCs and the evaluation of DID adequacy. Also, some technology-inclusive, risk-informed, and performance-based (TI-RIPB) methods and process will be applied to the FMR LBE selection. It is very important that use of the TI-RIPB process for selection of LBEs is endorsed by the NRC policies and compatible with the latest regulatory framework. This is critical to ensuring that the FMR design meets the regulatory requirements and can be licensed and operated safely. The FMR design team will also engage with the NRC staff early and frequently in the design process to ensure that the selection of LBEs is consistent with the regulatory requirements.

The followings are included in this report:

- An overview of the regulations and guidance to be considered during development of the LBE selection and classification approach
- Description of the process of LBE selection with a TI-RIPB approach
- Discussion of how probabilistic risk assessment (PRA) techniques will be used
- Summary and conclusions.

## 2. DESIGN FEATURES OF GA-EMS FMR

The FMR is a gas-cooled fast reactor (GFR), operating at system temperature range of 506 °C to 824 °C. It is a grid-capable power source with a gross electric output of ~44 MW. The reactor core uses helium coolant and uranium dioxide ($UO_2$) fuel pellets encapsulated in a silicon carbide (SiC) composite cladding, arranged in a triangular pitch, and forming a hexagonal fuel assembly.

The reactor core is an annular shape surrounded by solid reflector blocks of zirconium silicide ($Zr_3Si_2$) and graphite that preserve neutrons and enhance heat transfer. $Zr_3Si_2$ is a heavy reflector specifically developed for the GFR. This material is favorable in fast reactors to avoid power peaking around the core periphery from neutron thermalization.

Helium is chemically inert and will not aggravate an accident by contributing to any chemical or nuclear reaction. The use of helium as the coolant in combination with conventional fuel and effective neutron reflector offered enhanced neutronic and thermal efficiencies and several advanced safety characteristics such as efficient fuel utilization, high temperature operation, and inherently safe design that minimize the likelihood of accidents. For example, the helium coolant is intrinsically safe for it does not react with other materials or burn in air. The major systems and components are underground as illustrated in Figure 1.

The Power Conversion System (PCS) is a crucial component of the FMR power plant that converts the thermal energy generated by the reactor into electricity. The concept of the FMR PCS is similar to that for the gas turbine-modular helium reactor (GT-MHR). GA-EMS will develop the PCS of the FMR based on the previous experiences with the conceptual design of power conversion unit (PCU), leveraging the latest advancements in power conversion technology to optimize the efficiency and reliability of the PCS (i.e., PCU + generator system). The turbine-compressor-generator (TCG) are mounted on an inline vertical configuration. The generator is in a separate, connected vessel at the top of the PCU.

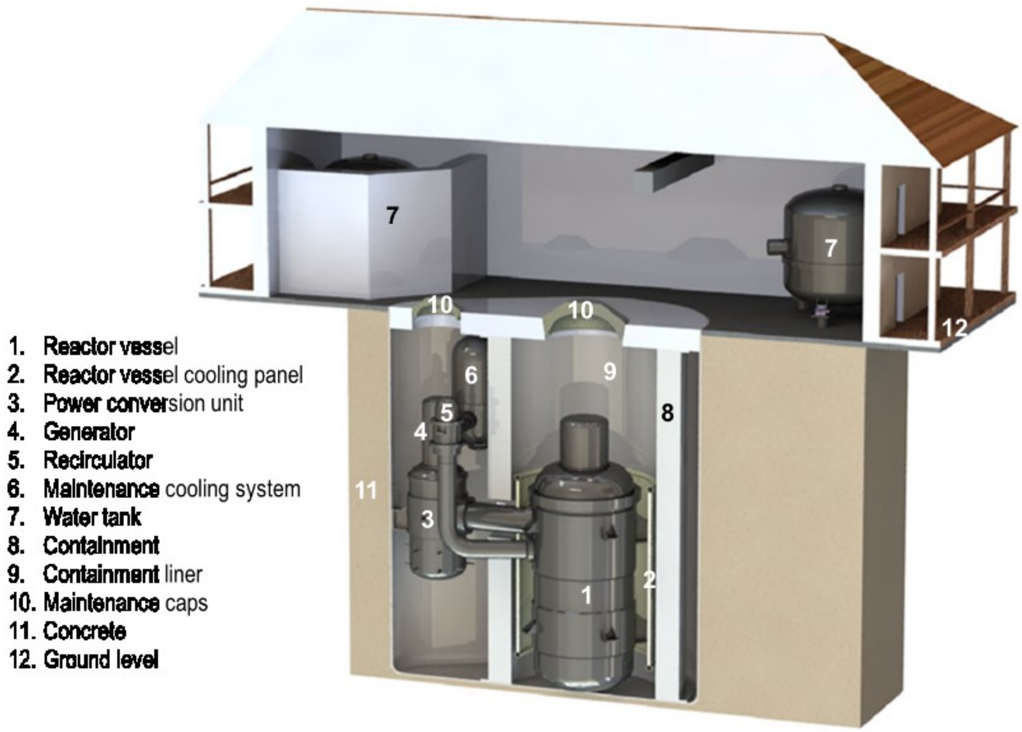


**Figure 1. FMR Nuclear Island Components**

One of the advanced design features of the FMR is its ability to passively remove decay heat from the core and vessel, regardless of whether helium is present. This is achieved through the implementation of a gravity-driven reactor vessel cooling system (RVCS). RVCS is always in operation [Reference 3] and continues passively removing the heat from the reactor vessel by natural circulation of water circulating in the RVCS loop. Unlike traditional gas-cooled reactors, which are typically packed with solid graphite, the FMR does not rely on conduction-cooldown. Instead, the passive safety of the core is primarily enhanced by the radiation heat transfer mechanism. For a rodded core like the FMR, the radiation heat transfer is the dominant heat transfer mechanism from the fuel rods to the surrounding solid structures, rather than conduction or convection.

Other design features, such as the large thermal margin, low power density, and annular core configuration, further enhance the passive safety of the core. Heat from the reactor vessel is transferred to the cooling panel of the RVCS through radiation. This system ensures that any decay heat generated by the core can be safely and efficiently removed, without the need for active cooling systems or other complex mechanisms. As a result, the FMR is able to offer exceptional levels of safety and reliability, making it an attractive option and a significant advancement in nuclear power generation technology.

# 3. REGULATORY REQUIREMENTS AND GUIDANCES

## 3.1. Federal Regulations

The following are the federal regulations that are relevant to the licensing basis for nuclear power plants.

*10 CFR Part 50:*

This regulation establishes the licensing requirements and standards for the operation of commercial nuclear power plants. It covers various aspects such as safety, security, radiation protection, emergency preparedness, and decommissioning. Part 50 sets forth comprehensive safety and security requirements for nuclear power plants to ensure the protection of workers, the public, and the environment. It covers various aspects such as design, construction, equipment, operations, maintenance, emergency preparedness, and radiation protection.

The regulation establishes standards for radiation protection to limit and control occupational and public exposure to radiation. It includes requirements for monitoring, record-keeping, training, and the implementation of measures to minimize radiation risks. Part 50 mandates the development and maintenance of emergency preparedness plans by license holders. These plans outline procedures and measures to be taken in the event of an accident or other emergency, and they are coordinated with state and local authorities.

- 10 CFR 50.34 (Contents of applications; technical information) describes the minimum information required for (a) preliminary safety analysis reports supporting applications for a construction permit and (b) final safety analysis reports supporting applications for operating licenses.
- 10 CFR Part 50 Appendix A provides the general design criteria for nuclear power plants, which include requirements related to reactor coolant systems, containment structures, emergency core cooling systems, and other important safety features.

*10 CFR Part 52:*

Although primarily focused on the licensing process, Part 52 is relevant to licensing basis events as it establishes requirements for the construction and operation of new nuclear power plants. It includes provisions for safety evaluations, emergency preparedness, and the development of site-specific design and construction criteria. This part governs the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities.

- Combined License (COL): The regulation introduces the concept of a Combined License, which combines the construction permit and operating license into a single license. This streamlines the licensing process by addressing both phases concurrently. 10 CFR 52.79, “Contents of applications; technical information in final safety analysis report,” describes the information to be included in final safety analysis reports supporting COLs.

- Early Site Permit (ESP): Part 52 allows for the issuance of an Early Site Permit, which authorizes site preparation activities before a specific reactor design is selected. This provides flexibility in planning for future nuclear power plants.
- Standard Design Certification: The NRC can issue a Standard Design Certification for a particular reactor design that meets established safety criteria. This allows subsequent applicants to reference the certified design in their license applications, reducing duplicative reviews. 10 CFR 52.47, “Contents of applications; technical information,” describes the information to be included in final safety analysis reports supporting applications for standard Design Certifications.
- Design Certification Rule: The regulation outlines the requirements for obtaining a Design Certification, including safety evaluations, technical specifications, and environmental impact assessments.
- Emergency Planning: The regulation requires license holders to develop and maintain emergency plans for responding to accidents and other incidents at nuclear power plants. These plans must be coordinated with state and local authorities.

### 3.2. Risk-Informed Performance-Based Regulation

The NRC's policy statement titled "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" was issued in 1995 [Reference 4] and played a crucial role in expanding the use of risk-informed practices by both the NRC and the industry. A year later, the NRC commissioners issued a staff requirement memorandum (COMSECY-96-061) [Reference 5] that focused on prioritizing regulatory efforts on licensee activities that posed the highest risk to the public, in order to accomplish the agency's principal mission in a cost-effective and efficient manner. This memo was supported by a white paper (SECY-98-144) [Reference 6] issued by the agency in 1998 that defined terms such as "risk informed" and "performance based" and provided expectations for the implementation of risk-informed, performance-based approaches.

In 2012, the NRC published a strategic vision document, NUREG-2150 [Reference 7], that outlined options for a more comprehensive, holistic, risk-informed, and performance-based regulatory approach. This document built upon the prior policies of the NRC for the use of risk-informed practices and set expectations for all areas of the agency's activities in a comprehensive manner.

More recently, in SECY-18-0060 [Reference 8], the NRC staff proposed the development of a technology-inclusive, performance-based regulation as an alternative approach for licensing non-




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light-water reactors. The staff also proposed transforming the review process to use risk insights to guide the scope, focus, and depth of a review.

A risk-informed approach to regulation by the NRC incorporates an assessment of safety significance or relative risk, ensuring that regulations are appropriate to their importance in protecting public health and safety. This approach considers risk insights along with other factors to establish requirements that focus on design and operational issues relevant to safety. It enhances the deterministic approach by considering a broader range of safety challenges, prioritizing them based on risk significance, and allowing for the consideration of additional resources and uncertainties. It also improves decision-making by testing the sensitivity of results to key assumptions.

Performance-based regulation focuses on achieving measurable outcomes without direction from the NRC specifying how they should be achieved. It establishes performance criteria based on risk insights, deterministic analyses, and historical performance, providing flexibility to licensees to meet these criteria in ways that encourage improvement. It emphasizes monitoring system performance and offers incentives for safety enhancement without immediate safety concerns arising from failure to meet performance criteria. Measurable parameters can be included in regulations or license conditions, allowing for monitoring and assessment of performance.

The NRC is recommending the addition of Part 53 (Risk-Informed, Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants) 10 CFR. The draft proposed rule in SECY-23-0021 offers a voluntary, performance-based alternative regulatory framework for licensing future commercial nuclear plants. In the context of this proposed rulemaking, future commercial nuclear plants, including non-light-water reactors (non-LWRs) and LWRs, would have the option to be licensed under Part 53. Applicants for these facilities would continue to have the option to be licensed under the existing requirements in 10 CFR Part 50 (Domestic Licensing of Production and Utilization Facilities) or 10 CFR Part 52 (Licenses, Certifications, and Approvals for Nuclear Power Plants).

In addition to providing a new Part 53, the draft proposed rule includes revisions to 10 CFR Part 26 (Fitness for Duty Programs) and 10 CFR Part 73 (Physical Protection of Plants and Materials) to address the possible attributes of future commercial nuclear plants. The draft proposed rule also includes conforming changes to other parts such as 10 CFR Parts 2 (Agency Rules of Practice and Procedure), 20 (Standards for Protection Against Radiation), 21 (Reporting of Defects and Noncompliance), 50, and 51 (Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions) among others.

### 3.3. NUREG-1860

The purpose of NUREG-1860 [Reference 9] is to establish the feasibility of developing a risk-informed and performance-based regulatory structure for the licensing of nuclear power plants. As such, this NUREG documents a "Framework" that provides an approach, scope and criteria




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that could be used to develop a set of requirements that would serve as an alternative to 10 CFR 50 for licensing nuclear power plants.

The objectives of the Framework include:

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

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

- Element 1: Goals and Expectations - These include the goals and expectations outlined in the Atomic Energy Act, emphasizing the adequate protection of public health and safety, the NRC's safety expectations based on the Safety Goal policy and the Quantitative Health Objectives (QHOs), and the NRC's security expectations for advanced reactors to employ enhanced safety margins and innovative approaches to fulfill their safety and security functions [Reference 10].
- Element 2: Defense-In-Depth - A core principle of the NRC's safety philosophy, DID is essential for addressing uncertainties associated with new technologies and design features, as well as uncertainties regarding safety challenges. Proper treatment of uncertainties is crucial in the licensing process of future reactors to ensure safety limits are met and the design is resilient against unanticipated factors.
- Element 3: Safety Fundamentals - This element outlines the process for translating high-level goals and expectations into specific requirements and regulations that implement safety, security, and preparedness expectations. Safety fundamentals are established through a DID approach, with protective strategies designed to ensure public health and safety with a high level of confidence. By conducting a top-down analysis of each protective strategy, appropriate requirements can be categorized and implemented to ensure the fulfillment of these strategies.

- Element 4: Licensing Basis - The goal is to incorporate risk insights throughout the regulatory structure, from the early stages of plant conception. The framework utilizes probabilistic criteria integrated with deterministic criteria based on plant specific considerations. In using a probabilistic process, confidence in the technical acceptability of the PRA becomes a key factor. The licensing basis criteria align with the Protective Strategies and support the NRC's defense-in-depth expectations, including compliance with the QHOs of the safety goals.

- Element 5: Integrated Process - This involves an integrated process for identifying requirements, starting with the examination of protective strategies. Each strategy is analyzed to identify potential threats or challenges that could lead to its failure, employing a logic tree and systems analysis. The principles of DID are applied to each strategy, leading to the identification of measures that can be incorporated into the requirements to prevent strategy failure. This systematic approach guides the selection of "topics" and the identification of specific requirements for each topic.

### 3.4. Licensing Modernization Project

The Licensing Modernization Project (LMP), led by Southern Company and cost-shared by the U.S. DOE and other industry participants, proposed changes to specific elements of the current licensing framework and a process for implementation of the proposals. These proposals will collectively lead to modernization and adaptation of the current licensing framework to support licensing of advanced non-LWRs. These proposals are intended to retain a high degree of nuclear safety, establish stable performance-based acceptance criteria, and enable near-term implementation of non-LWR design development, in support of national and industrial strategic objectives. The LMP objective is to support NRC efforts to develop regulatory guidance for licensing advanced non-LWR plants.

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

The new framework consists of elements including:

- Establishment of TI-RIPB licensing-basis event selection,




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- Classification of SSCs, and
- Establishment of predictable means to determine and preserve adequate DID.

These process steps are facilitated and informed by describing approaches and methods for: risk-informed decision making; the conduct and application of PRA 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.

### 3.5. Regulatory Foundation for Establishing Top-Level Regulatory Criteria (TLRC)

The aim of this section is to lay out criteria that establish limits for the frequencies or consequences of LBEs and LBE categories that must be considered when designing and operating a nuclear power plant to ensure public safety and to assess the adequacy of SSCs that perform safety functions during such events.

The following primary sources have been identified as containing criteria that establish limits on the risk or consequences of potential radiological releases from nuclear power plants in the U.S.

- Reactor Safety Goal Policy Statement (51 FR 28044) [Reference 10] - On August 4, 1986, the NRC adopted a safety goal policy for the operation of nuclear power reactors. The objective of this policy is to establish goals that broadly define an acceptable level of radiological risk. Two qualitative safety goals supported by two quantitative objectives were established. These two supporting objectives are based on the principle that nuclear risks should not be a significant addition to other societal risks.

  This policy limits public safety risk resulting from nuclear power plant operation. Limits are stated in the form of the maximum allowable risk of immediate death and the risk of delayed mortality from exposure to radiological releases of all types from nuclear power plants.

- 10 CFR Part 20, Standards for Protection against Radiation (Subpart C, Occupational Dose Limits) - The regulations promulgated under 10 CFR Part 20 establish standards for protection against ionizing radiation resulting from activities conducted under licenses issued by the NRC. Event sequences expected to occur within the plant lifetime, considering multiple reactor modules, are classified as AOOs. AOOs are evaluated against the dose limits of 10 CFR Part 20.

- 10 CFR Part 20, Standards for Protection against Radiation (Subpart D, Radiation Dose Limits for Individual Members of the Public) - These criteria (§20.1301) limit the dose consequences of releases associated with relatively high frequency events that occur as part of normal plant operations.




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- 10 CFR Part 50 Appendix I, Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low As Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents - This appendix provides explicit limits on doses from planned discharges that meet the NRC's definition of "As Low As Reasonably Achievable" (ALARA).
- 10 CFR Part 52 Subpart C, Combined Licenses - Under the provisions of 10 CFR §52.79, an application for a combined license must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for SSCs important to safety. This provides reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.
- 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operations - These standards provide the generally applicable exposure limits for members of the general public from all operations except transportation and disposal or storage of spent fuel associated with the generation of electrical power by nuclear power plants.
- 10 CFR Part 100, Reactor Site Criteria (Subpart B, Evaluation Factors for Stationary Power Reactor Site Applications on or After January 10, 1997) - §100.20 defines the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) of a nuclear reactor site and requires that the combination of the site and reactor located on that site be capable of meeting the dose and dose rate limits set forth in 10 CFR §50.34(a).
- 10 CFR §50.34(a)(1)(ii)(D), Contents of Applications, Technical Information - This section of the regulation specifies dose limits for evaluating the acceptance of the engineered safety features that are intended to mitigate the radiological consequences of accidents. These dose limits are consistent with those utilized in 10 CFR Part 100 for determining the extent of the EAB and Emergency Planning Zone (EPZ).

### 3.5.1. TLRC Related to Normal Operation and AOOs

The U.S. Environmental Protection Agency (EPA) is the agency given the authority to set generally applicable regulations governing the acceptable level of radiological exposure to members of the public.

Specifically, 40 CFR §190.10(a) states that the annual dose equivalent to a member of the general public from planned uranium fuel cycle operations shall be <25 mrem to the whole body, <75 mrem to the thyroid, and <25 mrem to any other organ. Portions of these exposure limits must be allocated to the various elements comprising the uranium fuel cycle (e.g., uranium mining and milling, fuel production, and reactor operations to produce electrical power). While the definition of 'uranium fuel cycle operations' specifically references the production of electric power by LWRs, the inclusion of the alternative term 'nuclear fuel cycle' in the definitions section of the regulation, as well as its formal title “Environmental Radiation Protection Standards for Nuclear Power Operations,” can be inferred to mean that it applies to other types of nuclear fuel as well as other types of reactors. Variances from these limits are allowed for unanticipated occurrences that still fall within the category of normal operations.

The NRC is the agency directly responsible for regulating the operation of nuclear power plants. As such, it is authorized to develop its own regulations, consistent with the requirements of 40 CFR Part 190, to ensure the health and safety of the general public. In exercising this responsibility, the NRC has promulgated a number of regulations that limit doses to the public from anticipated and unanticipated events during normal reactor operations.

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

10 CFR §50.34(b)(3) states that the means for controlling and limiting effluent releases and radiation exposures during operation shall be capable of meeting the requirements set forth in 10 CFR Part 20. 10 CFR §20.1301 requires that the total effective dose equivalent (TEDE) for a member of the public be limited to 100 mrem per year and 2 mrem in any 1 hour, in unrestricted areas. This regulation provides the applicable criteria for limiting dose to the general public from anticipated and unanticipated events associated with the normal operation of a nuclear power plant.

10 CFR Part 50, Appendix I, identifies dose and dose rate limits and limits on planned releases from the operation of nuclear power plant radwaste systems during normal operation, to maintain exposures ALARA. These criteria provide implementation guidance for applying the requirements of 10 CFR §50.34(a) and §50.36(a), for planned releases from the radwaste systems of nuclear power plants to the general environment to be ALARA. These ALARA limits are small fractions of the limits imposed by 10 CFR Part 20.

In setting its own quantitative limits on radiation exposure limits for the operation of nuclear power plants, the NRC has made use of the variance provided in 40 CFR §190.11 discussed earlier. The higher exposure limits set by 10 CFR Part 20 are associated with events still considered to lie within the regime of normal operations, but which are not considered to be ‘planned’ events as that term is used in 40 CFR Part 190.

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

### 3.5.2. TLRC Related to DBEs

10 CFR §50.34(a)(1) contains NRC’s regulations governing the design of new reactors and the means provided to protect against DBEs.

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

- An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem TEDE.
- An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from a postulated fission product release, would not receive a radiation dose in excess of 25 rem TEDE.

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

The regulations do not define DBAs in terms of their expected frequencies of occurrence, but 10 CFR §50.34(a)(1)(ii) articulates the expectation that the design, construction, and operation of nuclear power reactors will be such as to produce "an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products." No quantitative definition of the term 'extremely low probability' is provided in the regulation.

### 3.5.3. TLRC Related to BDBEs

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

The NRC's "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants" [Reference 11] states the Commission's intent to "*take all reasonable steps to reduce the chances of occurrence of a severe accident involving substantial damage to the reactor core and to mitigate the consequences of such an accident should one occur*." As noted earlier, this policy statement specifically addresses the Commission's intent to resolve safety issues associated with "accidents more severe than design basis accidents." This policy statement also makes the following points with respect to the design and licensing of new nuclear power plants:

- New plants are expected to achieve a higher standard of severe accident safety performance than existing plants.
- Innovative, cost-effective ways of achieving improved overall reliability for systems that prevent or mitigate the consequences of severe accidents are supported by the NRC.

- Analyses of events beyond the design basis should be as realistic as possible and make use of the insights provided by PRA.

In addition to its Severe Accident Policy, the NRC has issued NUREG-0880, “Safety Goals for Nuclear Power Plant Operation” [Reference 12] and the related policy statement entitled “Safety Goals for the Operations of Nuclear Power Plants” [Reference 10]. Two qualitative safety goals are used to express the NRC’s policy regarding the acceptable level of radiological risk from nuclear power plant operation as follows:

Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.

Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The following QHOs were identified as the basis for determining achievement of the above safety goals:

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

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

### 3.6. NEI 18-04

This NEI technical report [Reference 2] presents a TI-RIPB process for the selection of LBEs, safety classification of SSCs, and determination of DID adequacy for non-LWRs. This guidance provides an acceptable method for establishing the aforementioned topics as part of demonstrating a specific design provides reasonable assurance of adequate radiological protection. The process is focused on establishing guidance for advanced designs so license applicants can develop inputs that can be used to demonstrate compliance with applicable regulatory requirements. The process does not exempt any reactor designer from existing regulations but describes an approach to inform the safety design approach, which can then be applied to demonstrate compliance with the regulations applicable to a reactor design. The process described in this guidance document is a systematic and reproducible process for selecting LBEs, classifying SSCs, and determining the adequacy of DID in the design of nuclear power plants.

The process is risk-informed, which means it uses insights from systematic risk assessment in combination with structured prescriptive rules to address the uncertainties which are not addressed in the risk assessment. The approach provides reasonable assurance that adequate protection is provided for public radiological protection.

The process is also performance-based. The performance-based approach evaluates the effectiveness relative to realizing desired outcomes that are achieved by using quantifiable performance metrics for LBE frequencies and consequences and performance requirements for SSC capabilities to prevent and mitigate events. This is an alternative to a prescriptive approach specifying particular features, actions, or programmatic elements to be included in the design or process as the means for achieving desired objectives.

The processes covered in this guidance document are integrated and highly interdependent, starting with the process for the selection of LBEs. The outcomes from executing the processes in this guidance support developing a risk-informed and performance-based safety basis for the design. The process is also helpful in developing a safety-focused application for NRC review by systematically demonstrating the following:

- The selected LBEs adequately cover the range of hazards specific to the design and reflect the appropriate SSC failure modes.
- The LBEs are defined in terms of successes and failures of SSCs that perform safety functions modeled in the PRA, which are responsible for preventing and mitigating unplanned radiological releases from any source within the plant.
- The SSCs that perform the safety functions are collectively capable, reliable, diverse, and redundant across the layers of defense in the design.
- The design and programmatic features included in the licensing application demonstrate the philosophy of DID, and the outcomes of DID adequacy evaluations ensure adequate layers of defense.
- Plant capabilities and programmatic capabilities are reconciled based on risk-informed insights to provide reasonable assurance of adequate protection.
- The safety and risk significance of plant SSCs and programmatic controls included in applications are commensurate with their scope and level of detail.

### 3.7. RG 1.233

This regulatory guide (RG) [Reference 13] provides the NRC staff's guidance on using a TI-RIPB methodology to inform the licensing basis and content of applications for non-LWRs. This RG







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may be used by non-LWR applicants applying for permits, licenses, certifications, and approvals under 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities”, and 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants”.

The NRC staff has determined that the methods described in NEI 18-04 constitute one acceptable means to identify LBEs, classify SSCs, establish special treatments, identify programmatic controls, and assess DID for non-LWRs. These activities also define a methodology for applicants to identify and provide the appropriate level of information needed to satisfy parts of the regulatory requirements in 10 CFR 50.34, 10 CFR 52.47, 10 CFR 52.79, 10 CFR 52.137, and 10 CFR 52.157.

# 4. SELECTING AND EVALUATING LBES

## 4.1. Overview of Selecting LBEs

The selection of LBEs is a critical part of the licensing process for new nuclear power plants, as it ensures that the plant meets regulatory requirements and is safe for the public and the environment. The process for selecting LBEs for a reactor design should meet the following:

- Systematic and Reproducible: The LBE selection is a structured and consistent process that is involved with relevant stakeholders, such as regulatory authorities, industry experts, and the public. The process should be well-documented, clearly outlining the steps, criteria, and considerations used in the selection of LBEs. Consistent criteria are essential for evaluating events and should be systematically applied to all potential events, enabling a standardized and uniform analysis. The selection process should involve subject matter experts who possess the necessary knowledge and experience to assess events accurately. The selection process should be transparent, allowing for external scrutiny and verification. This transparency helps build trust and confidence in the chosen LBEs and the overall licensing framework.
- Sufficiently Complete: The process should consider all possible events that could impact the safety of the facility. It is necessary to conduct comprehensive analyses of potential hazards, threats, and accident scenarios that may arise during the operation of the facility. The selection process should identify and prioritize LBEs that are most relevant and significant in terms of their potential impact on the safety of the facility. Comprehensiveness in the selection process enhances the overall safety of the facility by allowing for a thorough understanding of the safety challenges and enabling implementation of appropriate measures to mitigate the effects of those events.
- Available for Timely Input to Design Decisions: The process should be completed in a timely manner, ensuring that the selected LBEs are considered in the facility's design decisions. It is important that the selection process should be initiated early in the design phase and completed before significant design decisions are made.




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- Reactor Technology-Inclusive: The process should be inclusive of the reactor technologies to ensure that the selected LBEs are appropriate for the specific type of reactor being designed.
- Risk-Informed and Performance-Based: The selection of LBEs should be based on risk-informed and performance-based criteria. "Risk-informed" approaches lie between the "risk-based" and purely deterministic approaches. By considering risk insights alongside other factors, this approach establishes requirements that focus attention on design and operational issues that are most critical to safety. The selecting process should take into account the frequency and consequences of potential events, as well as the facility's ability to mitigate their effects. By integrating risk insights, this approach enhances the effectiveness and efficiency of regulatory decision-making, ultimately leading to improved safety outcomes.
- Consistent with Applicable Regulatory Requirements: The process should be consistent with applicable regulatory requirements, including the NRC's safety requirements and guidance.

Selecting the licensing basis events for a nuclear reactor involves determining the set of accidents and scenarios that the reactor design and safety systems should be able to withstand. This selection process is crucial for ensuring the safety and regulatory compliance of the nuclear facility. However, there are several challenges associated with this task. Here are some of the key challenges:

- Complexity of the Reactor System: Nuclear reactors are highly complex systems with numerous interdependencies between components and processes. Identifying the full range of potential events and their consequences can be challenging due to the intricate nature of the reactor design.
- Uncertainty in Event Probability: Assessing the probability of various events can be difficult, especially for rare or low-probability scenarios. Limited historical data on nuclear accidents may make it challenging to accurately estimate the likelihood of specific events occurring.
- Regulatory Requirements: Nuclear reactors must comply with stringent regulatory standards imposed by the NRC. The challenge lies in aligning the selection of licensing basis events with these requirements while ensuring comprehensive coverage of all safety-related aspects.
- Adequacy of Design Basis: Selecting the appropriate licensing basis events necessitates determining the set of events that will test the safety systems and design limits of the reactor. It is crucial to ensure that the chosen events adequately represent the full

spectrum of potential hazards and challenges that the reactor might face during its operational lifetime.

- Deterministic vs. Probabilistic Approach: Deciding whether to use a deterministic or probabilistic approach for selecting licensing basis events is another challenge. A deterministic approach focuses on identifying specific event categories and their consequences, while a probabilistic approach considers the likelihood of events based on statistical analyses. Balancing these two approaches and integrating them effectively can be a challenge.

The TI-RIPB approach for selecting LBEs is a systematic and comprehensive method aimed at ensuring that the appropriate set of events that could impact the safety of a nuclear reactor design is thoroughly considered. By employing this approach, the goal is to identify and define the risk-significant events that must be addressed in the design and licensing process. The TI-RIPB approach recognizes the significance of incorporating risk insights into decision-making processes related to design and licensing. It emphasizes the need for a comprehensive evaluation of LBEs, which are events or conditions that have the potential to challenge the safety systems of the reactor. These LBEs are categorized and evaluated individually, allowing for a detailed assessment of the performance of SSCs in response to each initiating event.

By evaluating LBEs individually, the TI-RIPB approach ensures that the safety systems of the reactor are capable of effectively responding to specific events or conditions. This evaluation process takes into account regulatory limits and criteria, ensuring compliance with established safety standards. Additionally, the approach considers the integrated risk of the reactor design, aligning with the safety goals defined by the NRC.

Through the comprehensive evaluation of LBEs, the TI-RIPB approach aims to provide a comprehensive understanding of the risk profile associated with a nuclear reactor design. It ensures that safety considerations are integrated into the design and licensing decisions, enabling the identification and implementation of necessary measures to mitigate potential risks effectively. By adhering to this approach, the design and licensing processes can be guided by a thorough assessment of risk and the assurance that the safety systems in place are capable of fulfilling their intended functions in the face of challenging events or conditions.

Ensuring the completeness of LBE selection is crucial for developing a comprehensive safety basis. The following are some key steps and considerations to ensure the thoroughness and completeness of LBE selection:

- Identify Applicable Regulations and Requirements: Review the applicable regulations, standards, guidelines, and licensing requirements that govern the facility or system under consideration. Understand the specific events and scenarios that need to be addressed in the safety basis.





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- Conduct a Systematic Analysis: Perform a systematic analysis of the facility, system, or process to identify potential hazards, failure modes, and initiating events. Consider both internal and external events that could impact the safety of the facility.
- Utilize Risk Assessment Techniques: Apply various risk assessment techniques to identify potential LBEs. These techniques help identify the combinations of events, failures, or conditions that could lead to hazardous situations.
- Consider Design Basis and Design Envelope: Review the facility's design basis and design envelope, which specify the intended operating conditions, design limitations, and design margins. Ensure that the selected LBEs are consistent with the design basis and encompass a broad range of credible events that could challenge the safety of the facility.
- Involve Multidisciplinary Experts: Engage a team of multidisciplinary experts, including engineers, operators, maintenance personnel, safety professionals, and subject matter experts, to ensure a comprehensive understanding of the facility and its potential hazards. Collaborative input from diverse perspectives helps identify LBEs that might be overlooked by a single individual or discipline.
- Consider Different Initiating Events: Consider a wide range of initiating events, including equipment failures, process upsets, natural phenomena, human errors, external hazards, and combinations thereof. Ensure that the selected LBEs encompass a spectrum of events that reflect both common and rare occurrences.
- Review and Refine LBE Selection: Regularly review and refine the list of selected LBEs throughout the safety basis development process. Incorporate feedback and input from stakeholders, regulatory authorities, and independent reviewers to validate the completeness of the selection.
- Document and Justify LBE Selection: Document the rationale and justification for the selected LBEs, including the analysis methods used, the criteria for inclusion, and the justifications for excluding certain events. Clearly articulate how the selected LBEs encompass a comprehensive range of credible hazards and address the requirements set by regulations and standards.
- Continuous Improvement: Recognize that the process of LBE selection is not a one-time exercise but an ongoing effort. Continuously monitor and reassess the completeness of the safety basis, considering changes in regulations, technology, operational experience, and lessons learned from industry-wide events.

### 4.2. Event Categorization

There are four categories of LBEs:

- Anticipated Operational Occurrences (AOOs), which encompass normal operation and planned and anticipated events whose frequencies exceed $10^{-2}$/plant-year where a plant may be comprised of one or more reactor modules. The radiological doses from AOOs are required to meet normal operation public dose requirements. AOOs are utilized to set operating evaluation criteria for normal operation modes and states.
- Design Basis Events (DBEs), which encompass unplanned off-normal events not expected in the plant's lifetime whose frequencies are in the range of $10^{-4}$ to $10^{-2}$/plant-year, but which might occur in the lifetimes of a fleet of plants. The radiological doses from DBEs are required to meet accident public dose requirements. DBEs are the basis for the design, construction, and operation of the structures, systems, and components (SSCs) during accidents.
- Beyond Design Basis Events (BDBEs), which are rare off-normal events whose frequencies range from $5x10^{-7}$/plant-year to $10^{-4}$/plant-year. BDBEs are evaluated to ensure that they do not pose an unacceptable risk to the public and to provide input to the selection of DBAs.
- Design Basis Accidents (DBAs). The DBAs for Chapter 15, "Accident Analyses," of the license application are prescriptively derived from the DBEs by assuming that only SSCs classified as safety-related are available to mitigate the consequences. The public consequences of DBAs are based on mechanistic source terms and are conservatively calculated. The upper 95% conservative estimate of the dose of each DBA must meet the 10 CFR §50.34 consequence limit at the EAB. The DBAs often correspond to event sequences modeled in the PRA with extremely low frequencies.

### 4.2.1. AOO Region

AOOs refer to specific conditions of plant operation that are expected to occur one or more times throughout the lifespan of a nuclear facility. These conditions are considered as part of the design and operational planning processes to ensure that the facility is adequately prepared to handle them. A conservative value of $1x10^{-2}$ per plant year is used as a lower bound to establish the AOO region.

The NRC imposes strict requirements on nuclear facilities to design and operate in a manner that can accommodate these anticipated operational occurrences. The objective is to prevent these occurrences from escalating into more serious events that could pose a risk to the safety of workers, the general public, and the environment. By incorporating measures to address AOOs, nuclear facilities can maintain a high level of safety during normal operations. This emphasis on safety is essential to protect the well-being of individuals and to prevent any adverse impacts on the surrounding environment.

The requirement to accommodate anticipated operational occurrences serves as a proactive approach to risk management. By identifying and considering these potential events during the




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design and operational stages, nuclear facilities can implement appropriate engineering controls, safety systems, and operational procedures to mitigate their consequences. This proactive stance ensures that the facility remains in a safe and controlled state even when faced with expected operational occurrences.

The criteria outlined in 10 CFR Part 20 govern the management of releases of radioactive material during normal reactor operations, including AOOs. 10 CFR Part 20 provides specific numerical guidance to ensure that these releases are maintained ALARA, which means minimizing radiation exposure to individuals and keeping it at the lowest feasible levels. A key parameter defined by 10 CFR Part 20 is the TEDE, which quantifies the total radiation dose received by an individual from various sources of radiation exposure. TEDE takes into account external radiation exposure from the environment as well as internal exposure from the inhalation or ingestion of radioactive materials. 10 CFR Part 20 sets an upper limit for TEDE to ensure that the doses received by individuals in unrestricted areas surrounding the facility are within acceptable limits. Specifically, the standard sets the total effective dose equivalent at the EAB to be no greater than 100 millirem (mrem). The EAB is the designated area surrounding the reactor where the licensee holds authority over all activities, including the power to exclude or remove personnel and property.

The EAB is expected to coincide with the boundary of the Controlled Area, which is an area within the facility where access is restricted, and radiation levels are controlled. By maintaining TEDE at or below 100 mrem at the EAB, the aim is to protect individuals who may be present in the unrestricted area surrounding the reactor from excessive radiation exposure.

Adhering to the criteria outlined in 10 CFR Part 20 ensures that reactor licensees take appropriate measures to monitor and control radiation releases, implement shielding and containment strategies, and manage worker and public exposure to radiation. By applying these regulations, the nuclear industry maintains a commitment to safety, minimizing the potential health risks associated with radiation exposure and promoting a secure operating environment for both workers and the public.

#### 4.2.2. DBE Region

The DBE region encompasses releases of radioactive material that are not expected to occur during the operational lifespan of a single nuclear power plant. However, these events may be encountered during the operational lifespan of a population of nuclear power plants with similar designs. To establish the lower bound for this region, a conservative value of $1 \times 10^{-4}$ per plant-year is used.

The purpose of defining the DBE region and its associated frequency is to ensure that the NRC's Safety Goal of QHOs are met with an appropriate margin of safety. It is not necessary to require compliance with the criteria specified in 10 CFR §50.34 at frequencies lower than $10^{-4}$ per plant-year, as this lower frequency already provides sufficient assurance of meeting the safety goals.

The DBE region plays an important role in ensuring the safety of nuclear power plants and protecting the surrounding population from potential accidental releases of radioactive material. Within this region, specific criteria and guidelines are established to determine the level of protection required and to assess the potential risks associated with such events. One of the key aspects of the DBE region is the consideration of dose limits, specifically the TEDE criterion. In accordance with 10 CFR §50.34a, the TEDE criterion sets a quantitative dose guidance of 25 rem for accidental releases. This criterion serves as a crucial factor in siting a nuclear power plant, as it ensures that the surrounding population is adequately protected from radiation exposure. The 25 rem TEDE criterion provides a conservative threshold that considers potential uncertainties and factors in a margin of safety to protect public health.

The combination of selected frequency limits and dose limits within the DBE region is designed to align with the NRC’s Safety Goal QHOs. These objectives aim to minimize the individual risk of latent cancer fatality resulting from nuclear power plant operations. By adhering to the established frequency and dose limits, the DBE region provides a high level of confidence that the NRC's safety goals will be met with significant margins of safety.

### 4.2.3. BDBE Region

The BDBE region represents a category of events that are highly improbable and not expected to occur during the operational lifespan of a large fleet of nuclear power plants. These events are characterized by their extremely low occurrence probabilities. The purpose of considering the BDBE region is to ensure that the risk to the public from these low probability events remains at an acceptable level.

Unlike the DBE region, the BDBE region focuses on events that have a mean occurrence frequency below the lower limit set for the DBE region, typically $1 \times 10^{-4}$ per plant-year. While the specific events within the BDBE region may vary, they generally involve extreme or severe conditions that are beyond the scope of what is considered in the DBE region. Examples of BDBEs could include rare natural disasters, extreme external hazards, or severe beyond-design accidents that are not accounted for in the design basis of nuclear power plants. These events may have significant potential consequences, but their occurrence is considered highly improbable. These events are considered highly unlikely, and their occurrence probabilities fall well below the threshold established for the DBE region.

The consideration of the BDBE region is essential to ensure that the risk to the public remains acceptable even under extreme and unlikely scenarios. By assessing and addressing the potential risks associated with BDBEs, nuclear power plant operators, regulatory authorities, and other stakeholders can demonstrate that appropriate measures are in place to safeguard public safety.

It is important to note that the BDBE region goes beyond the normal scope of design basis considerations and requires a more comprehensive assessment of potential risks. The focus




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shifts from deterministic analysis to probabilistic risk assessment methodologies, which take into account the low probability but high-consequence events. These assessments involve complex analyses, modeling, and simulations to evaluate the potential impacts of BDBEs on the safety of nuclear power plants and the surrounding environment.

The consideration of the BDBE region contributes to the overall safety culture and regulatory framework of the nuclear industry. It ensures that measures are in place to address even the most unlikely events and provides an additional layer of protection for the public and the environment. By including the BDBE region in the overall risk assessment process, the nuclear industry demonstrates its commitment to safety and continuous improvement.

The frequency cutoff implicit in the acute fatality risk goal, as defined in NUREG-0880, "Safety Goals for Nuclear Power Plant Operation" [Reference 12], serves as the lower frequency boundary for the BDBE region. NUREG-0880 acknowledges that the individual mortality risk of prompt fatality in the United States is approximately $5 \times 10^{-4}$ per year for all accidental causes of death. This mortality risk design objective aims to limit the increase in an individual's annual risk of accidental death to 0.1% of $5 \times 10^{-4}$, which corresponds to an incremental increase of no more than $5 \times 10^{-7}$ per year. By adhering to this objective, it can be ensured that even in scenarios or sets of scenarios with frequencies at or below this value, the individual risk contributions would still align with the safety goal, regardless of the magnitude of the consequences, while maintaining a significant residual margin of safety.

The consideration of the acute fatality risk goal in establishing the BDBE region reflects a proactive approach to safety and risk management. It acknowledges that while the occurrence of events within the BDBE region is unlikely, the potential consequences can be significant. By defining a frequency cutoff that aligns with the incremental risk increase objective, regulatory authorities and industry stakeholders can prioritize the evaluation and mitigation of events with frequencies below this threshold. This approach ensures that appropriate measures are in place to minimize risks, protect public health and safety, and prevent adverse impacts on the environment.

The BDBE region serves as an additional layer of protection beyond the design basis of nuclear power plants. It requires a comprehensive analysis and assessment of potential scenarios with frequencies below the established cutoff. The analysis involves utilizing advanced probabilistic risk assessment methodologies, sophisticated modeling techniques, and thorough evaluations of potential hazards. Through this comprehensive analysis, the industry can identify and address potential risks, implement appropriate safety measures, and ensure that the risks associated with low-frequency events remain at an acceptable level.

### 4.3. Frequency - Consequence Curve

The frequency-consequence (F-C) evaluation criterion is a useful tool for assessing and managing risks associated with nuclear power plants. It provides a structured approach for





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evaluating hazards and identifying appropriate risk mitigation measures. The F-C evaluation criterion is a risk assessment approach used to evaluate the likelihood and potential impact of hazards that could result from the operation of nuclear systems. This approach is derived from the TLRC, which outlines the regulatory requirements for ensuring the safety of certain operations. The F-C curve provides an acceptable limit in terms of the frequency of potential accidents and their associated consequences. The objective of the F-C curve is to establish the licensing basis, i.e., identify the event sequences that the design and operation of the plant need to be able to mitigate. The objective involves establishing criteria that define the acceptable frequencies for different levels of consequences.

Acceptable offsite dose evaluation criteria on the event sequence consequences for the LBE categories are defined by a frequency-consequence evaluation criteria derived from TLRC. Key elements of the TLRC used to develop the frequency-consequence criteria include:

The F-C Target for lower frequency AOOs at frequencies of $10^{-1}$/plant-year down to $10^{-2}$/plant-year are set at a reference value of 1 rem corresponding with the EPA Protective Action Guide (PAG) limits [Reference 14] and consistent with the goal of avoiding the need for offsite emergency response for any AOO. It is expected that many LBEs will not result in the release any radioactive material, and the identification of plant capabilities to prevent such releases is a factor considered in the formulation of SSC safety classification and performance requirements.

The F-C Target for DBEs range from 1 rem at $10^{-2}$/plant-year to 25 rem at $10^{-4}$/plant-year with the dose calculated at the EAB for the 30-day period following the onset of the release. This aligns the lowest frequency DBEs to the limits in 10 CFR 50.34 and provides continuity to the lower end of the AOO criteria. A straight line on the log-log plot connects these criteria. The identification of plant capabilities to prevent releases is a factor considered in the formulation of SSC safety classification and performance requirements as discussed more fully in the section below on SSC safety classification. It is expected that many LBEs will not release any radioactive material.

The F-C Target for the BDBEs range from 25 rem at $10^{-4}$/plant-year to 750 rem at $5\times10^{-7}$/plant-year to ensure that the QHO for early health effects is not exceeded for individual BDBEs.

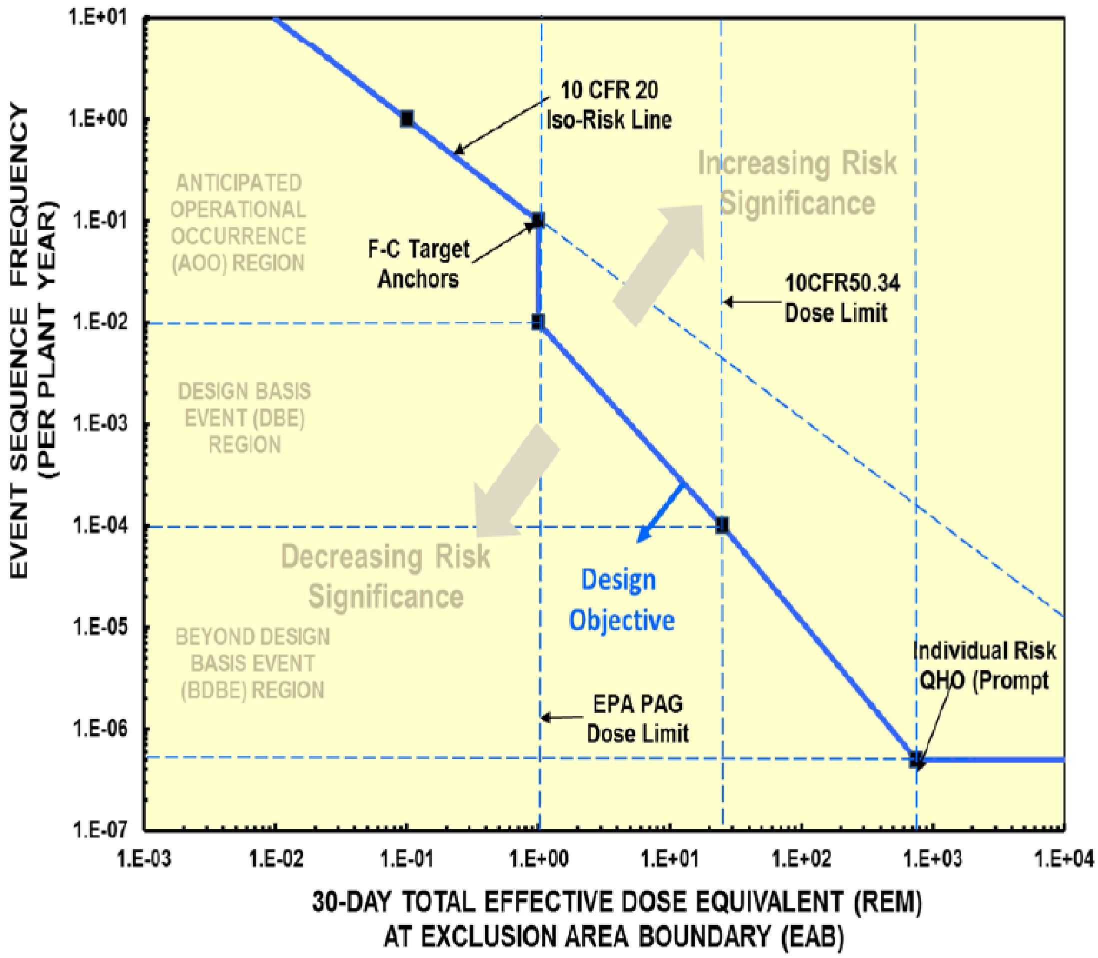


**Figure 2. Frequency – Consequence Curve**

### 4.4. TI-RIPB Approach to Selecting LBEs

A TI-RIPB process for the selection of LBEs is designed to be comprehensive and takes into account the latest technologies, risks, and performance criteria. The process is an approach used to identify and evaluate potential events that could affect the safety of a nuclear facility. The process is implemented in the following LBE selection tasks:

*Task 1. Propose Initial List of LBEs*

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

*Task 2. Design Development and Analysis*




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

*Task 3. PRA Development/Update*

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

*Task 4. Identify/Revise List of AOOs, DBEs, and BDBEs*

The event sequences modeled and evaluated in the PRA are grouped into accident families each having a similar initiating event, challenge to the plant safety functions, plant response, and mechanistic source term if there is a release.

*Task 5. Select/Revise Safety-Related SSCs*

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

*Task 6. Select DBAs*

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

*Task 7. Perform LBE Evaluations*

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

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

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

*Task 7b. Evaluate Integrated Plant Risk*

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

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

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

*Task 7c. Evaluate risk significance of Barriers and SSCs*

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

*Task 7d. Perform Deterministic Safety Analyses against 10 CFR 50.34*

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

*Task 7e. Risk-Informed, Performance-Based Evaluation of Defense-in-Depth*

In this task, the definition and evaluation of LBEs will be used to support a TI-RIPB evaluation of defense-in-depth. This task involves the identification of key sources of uncertainty, characterization of safety margins, and evaluation against defense-in-depth criteria that are the subject of a companion white paper to be developed in the LMP as a future deliverable.

*Task 8. Decide on Completion of Design/LBE Development*

The purpose of this task is to make a decision as to whether additional design development is needed to select the LBEs, either to proceed to the next logical stage of design or to incorporate feedback from the LBE evaluation that design improvements should be considered. Such design improvements could be motivated by a desire to increase margins against the F-C criterion, 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.

### 4.5. PRA Application to LBE Selection

PRA is a systematic and comprehensive analysis that quantifies the probabilities and consequences of potential events, including initiating events, accident sequences, and their associated risks. It provides insights into the overall risk profile of the plant and helps in identifying important events that need to be considered for licensing purposes.

The PRA will be used to evaluate the safety characteristics of the preliminary design and to provide a structured framework from which the initial set of LBEs will be risk-informed. In addition, engineering judgment and utilization of relevant industry experience will continue to be used to ensure that LBE selection and classification is complete.

As a powerful tool in the development of comprehensive safety bases for facilities, PRA aid in the design and evaluation of events included in the licensing basis. By leveraging the rigorous nature of PRA, facilities can maximize the probability of establishing a comprehensive safety basis that accounts for various potential risks.

Traditionally, safety evaluations have primarily relied on deterministic approaches that focus on single-point failures or worst-case scenarios. However, these approaches may overlook the systemic and interactive nature of risks. In contrast, PRA provides a rigorous approach to safety assessment, considering a wide range of potential events and their associated probabilities. PRA involves a systematic analysis of events and their likelihood of occurrence, enabling a quantitative understanding of risks. It combines knowledge from various disciplines, including engineering, statistics, and human factors, to develop a comprehensive model of a facility's risk profile. The process encompasses event identification, event frequency analysis, consequence assessment, and risk quantification.







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One of the significant advantages of PRA is its ability to evaluate the comprehensive performance of facility designs. Traditional deterministic approaches often focus on individual components or events, neglecting potential interactions and cascading effects. In contrast, PRA considers the interdependencies among systems, components, and human factors. This comprehensive evaluation helps identify vulnerabilities, anticipate potential failures, and prioritize risk mitigation strategies.

PRA facilitates risk-informed decision-making by quantifying risks associated with various design options, operational procedures, and mitigation strategies. It allows stakeholders to compare the potential benefits and trade-offs of different safety measures, optimizing resource allocation and regulatory requirements.

Using a PRA to aid in the development of events that are included in the licensing basis maximizes the probability of establishing a comprehensive safety basis. By its nature, PRA development is a rigorous process that considers the comprehensive performance of the facility design.

Probabilistic methods for event selection, SSC safety classification, special treatment identification, and integration of defense-in-depth strategies will seek to optimize the safety characteristics of the reactor design. The PRA provides a rational approach for identifying, understanding, and addressing uncertainties.

The PRA will systematically enumerate event sequences and assesses the frequency and consequence of each event sequence. Event initiators will include internal, common cause, and external events.

An event family is defined as a collection of event sequences that similarly challenge plant safety functions. The SSCs that perform the safety functions have a similar response, and the sequences have similar consequences. Event family grouping facilitates selection of LBEs from many individual events.

For example, the PRA will likely include variations of helium pressure boundary breaks in terms of break size and location relative to the core and different compartments in the reactor building. A particular family of helium pressure boundary breaks combines many individual break locations and sizes found to have similar plant response to facilitate common analysis and consistent application of required preventive and mitigative functions.

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

The NRC is expecting greater emphasis on the use of risk information, provided there is sufficient understanding of plant and fuel performance, and deterministic engineering judgment is used to bound uncertainties. The need for "sufficient understanding of plant and fuel performance" will be addressed by such topics as fuel qualification, high temperature materials, and mechanistic source terms.

In SECY-03-0047 [Reference 15], the staff recommended the use of a probabilistic approach for the development of the licensing basis (e.g., selection of events to be considered in the design, safety classification). The Commission approved the staff's recommendation in a June 26, 2003, Staff Requirements Memorandum (SRM). However, the details of how this was to be done were yet to be developed. The Framework (in Chapter 6) proposes an approach for implementing this recommendation. This approach relies on a full scope design-specific PRA and uses a probabilistic approach for:

- establishing frequency ranges to select and categorize events which need to be considered in the licensing basis,
- selecting LBEs and establishing plant design features,
- establishing LBE acceptance criteria as a function of the frequency of event scenarios,
- classifying certain SSCs as safety significant,
- replacing the single failure criterion, where practical, and
- establishing security performance standards.

Because the licensing basis is risk-derived, the approach also requires (1) maintaining the PRA up to date over the life of the reactor, and (2) continually reassessing plant risk, licensing basis events, safety classification, etc., using actual operating experience to determine if the plant licensing basis remains valid. This will involve feeding the results from the updated PRA back into plant operation and, where the licensing basis is affected, reporting the results to NRC.

The event sequences to be considered in the design are those with a frequency of $10^{-7}$/plant-year or larger. As such, some of the low frequency event sequences may include severe accidents. This would represent an extension of the licensing basis from one defined by traditional anticipated operational occurrences and DBAs, to one including accidents beyond traditional DBAs.

Event Sequence in PRA refers to the chronological order of events that are considered in the analysis to assess the risk associated with a particular system or process. PRA is a systematic methodology used to quantify and analyze the potential risks and vulnerabilities of complex systems, such as nuclear power plants, chemical plants, or transportation systems. The event sequence in PRA typically involves the following key elements:

Initiating Event: The initiating event is the initial event or condition that triggers the sequence of events. It can be an external event, equipment failure, human error, or a combination of factors.

- Failure Events: Failure events refer to the subsequent failures or malfunctions that occur in the system following the initiating event. These failures can include equipment failures, structural failures, or human errors that may lead to adverse consequences.
- Event Dependencies: Event dependencies represent the relationships between different events within the system. Dependencies can include events that are influenced by the outcome of previous events or events that are independent of each other.
- Safety Systems and Actions: Safety systems and actions are the measures and procedures in place to mitigate or prevent adverse consequences. These can include safety systems, emergency response actions, operator actions, or automatic shutdown systems.
- Consequences: Consequences are the outcomes or results of the events in the sequence. These can include various types of consequences, such as equipment damage, releases of hazardous materials, injuries to personnel, or environmental impacts.
- Success or Failure of Mitigation Measures: The effectiveness of the safety systems and actions in mitigating the consequences is assessed. This includes evaluating the success or failure of the implemented mitigation measures in preventing or reducing the severity of the consequences.

By analyzing the event sequence in PRA, it is possible to identify the potential pathways and scenarios that can lead to accidents or undesired events. The analysis helps in quantifying the likelihood of each event and estimating the associated risks. This information is then used to prioritize risk-reduction measures and improve the overall safety and reliability of the system.

While PRA offers significant benefits, its implementation in licensing processes requires careful consideration of several challenges. These challenges include data availability, model complexity, uncertainty analysis, and integration with existing regulatory frameworks. Collaborative efforts between facility operators, regulators, and PRA experts are essential to address these challenges effectively.

The level of PRA required for LBE selection can vary depending on several factors, including regulatory requirements, the complexity of the nuclear power plant, and the level of risk associated with the plant's operations. The extent of PRA needed for LBE selection is typically determined by regulatory authorities and industry standards.

In some cases, a full-scope PRA, which covers a wide range of initiating events, accident sequences, and associated probabilities, may be required as part of the licensing process. This type of PRA involves detailed modeling and analysis of plant systems, component reliability, operator actions, and other relevant factors to assess the plant's safety performance.

However, in other cases, a more focused or limited-scope PRA may be sufficient for LBE selection. This could involve analyzing specific events or accident scenarios that are deemed critical or have the highest potential impact on plant safety.

The specific requirements for PRA in LBE selection should be determined based on the applicable regulatory framework and guidance documents provided by the regulatory authorities. These documents typically outline the expectations for PRA quality, scope, methodology, and level of detail needed for licensing purposes.

It is important to consult the relevant regulatory authority or licensing body to obtain specific information regarding the PRA requirements for LBE selection for a particular nuclear power plant project.

### 4.6. Initiating Events

The identification of initiating events (IEs) is a crucial step in the safety assessment of nuclear power plants. It serves as the foundation for determining events that could lead to undesirable consequences and assessing the overall risk of the plant.

The process of identifying IEs requires an approach that utilizes multiple methods. By employing various tools and perspectives, confidence can be increased in producing a comprehensive list of IEs, ensuring that all foreseeable events are reasonably captured. Internal hazards, including internal events, internal floods, and internal fires, as well as external hazards such as seismic events, high winds, and external floods, must be considered during the identification process. The American Society of Mechanical Engineers (ASME)/ American Nuclear Society (ANS) non-LWR PRA standard [Reference 16] and industry studies like NUREG/CR-2300 [Reference 17], NUREG-1407 [Reference 18], International Atomic Energy Agency (IAEA) SSG-3 [Reference 19], and Electric Power Research Institute (EPRI) 1022997 [Reference 20] provide valuable guidance and insights into the potential hazards to be considered.

*Iterative Nature of Initiating Event Identification*

The process of identifying IEs is iterative and evolves with the design development process. It starts with a conceptual design and continues as the design matures and the understanding of plant operations improves. This iterative approach ensures that the list of IEs is refined and updated to reflect the changing design and operational conditions. The identification process should not be considered a one-time activity but rather a continuous effort throughout the plant's life cycle.

*Consideration of Design-Specific Factors*

To ensure a comprehensive and exhaustive identification of IEs, a thorough systematic search using appropriate methods is necessary. While industry standards and guidelines provide a preliminary list of hazards, it is essential to account for design-specific factors that may not be explicitly listed. Therefore, a comprehensive effort must be undertaken to identify and evaluate IEs specific to the design and operational characteristics of the nuclear power plant.

The identification of IEs is a critical step in the safety assessment of nuclear power plants. A blended and robust approach, utilizing multiple methods, ensures the completeness and accuracy of the identified initiating events. The iterative nature of the identification process, coupled with consideration of design-specific factors, allows for continuous refinement and updating of the list of initiating events throughout the plant's life cycle. By conducting a comprehensive and systematic search, nuclear power plant operators and regulators can confidently assess the risks associated with IEs and implement appropriate measures to enhance safety and reliability.

The major categories of IEs that can be identified for FMR reactors include:

1) Transient events with no reactivity addition - These are events where there is a change in the operating conditions of the reactor, but no additional reactivity is introduced. The following specific events are considered as examples:

    a. Spurious reactor trip: This refers to an unplanned and unexpected shutdown of the reactor due to a false signal or malfunction of the reactor trip system. It involves the automatic insertion of control rods to halt the chain reaction and bring the reactor to a safe shutdown state.

    b. Power turbine generator trip: This event occurs when the turbine connected to the reactor's power generator trips or shuts down unexpectedly. It can lead to a sudden reduction or loss of electrical power output from the reactor.

    c. Loss of offsite power: This refers to the loss of external electrical power supply to the reactor. It can result from various factors such as grid failures, severe weather events, or other external disturbances. The loss of offsite power requires the reactor to rely on alternative power sources or emergency backup systems to maintain safe operation.

2) Transient events with reactivity addition - These are events where additional reactivity is introduced.

    a. Control rod withdrawal: This event involves the partial or complete withdrawal of control rods from the reactor core.

    b. Control rod ejection: This event refers to the unintended and rapid expulsion or ejection of a control rod from the reactor core. Control rod ejection can occur due to mechanical malfunctions, structural failures, or other operational issues. It is an abnormal and potentially hazardous situation because the sudden removal of a control rod can lead to a rapid increase in reactivity and power, potentially exceeding safe operating limits.

3) Leaks or breaks in reactor coolant pressure boundary (RCPB)

a. Slow leaks in RCPB (<10 mm): This category refers to small and gradual leaks that may result from issues such as degraded seals, gaskets, or small cracks. Although the leaks are relatively small, they need to be addressed to prevent further deterioration and potential hazards.

b. Moderate leaks in RCPB (<100 mm): This category includes leaks in the RCPB that are larger than the previously mentioned slow leaks. Moderate leaks can occur due to more significant cracks, breaches, or failures in the reactor coolant pressure boundary. They pose a greater concern than slow leaks and require prompt attention to maintain the integrity of the RCPB and prevent any safety risks.

c. Large leaks in RCPB (>100 mm): This category involves leaks in the RCPB that can result from severe structural failures, ruptures, or breaches in the reactor coolant pressure boundary. These leaks can lead to significant loss of coolant, compromising the integrity and functionality of the reactor system. Large leaks require immediate response and mitigation to prevent further damage and ensure the safety of the reactor.

d. Breaks in RCPB resulting in water ingress to the core: This category refers to specific breaks or breaches in the reactor coolant pressure boundary that allow water from outside the core to enter the reactor core. This situation is particularly concerning because it can impact the reactor's cooling system and control mechanisms. Water ingress to the core can disrupt the normal operation of the reactor, potentially leading to overheating, fuel damage, or other safety issues.

4) Leaks or breaks in reactor cooling system to heat exchangers

a. Leaks or breaks in precooler: A leak or break in the precooler refers to a breach in its integrity, allowing the coolant to escape or causing a loss of coolant.

b. Leaks or breaks in intercooler: A leak or break in the Intercooler indicates a failure or damage to this component, resulting in coolant leakage or loss. This can disrupt the heat exchange process, impacting the overall efficiency of heat transfer.

c. Leaks or breaks in residual heat removal (RHR) heat exchanger: The RHR heat exchanger is a crucial component of the system responsible for removing residual heat from the reactor during normal and emergency conditions. A leak or break in the RHR heat exchanger refers to a breach in its structure, allowing the coolant or working fluid to escape or causing a loss of functionality.

The general categories of causes of an IE that would be considered when conducting a full scope PRA include the following:

1. Internal events: These are events that are caused by equipment failures or operator errors that occur within the system being analyzed. Examples of internal events include

malfunctions, component failures, human errors, or any other event that originates from within the system boundaries.

2. Internal plant hazard events: This category refers to specific types of hazards that can arise within the plant or system being assessed. It includes events such as internal fires, internal floods, or equipment-generated missiles. Internal fires can be caused by ignition sources within the plant, while internal floods may result from leaks or failures in the water management systems. Equipment-generated missiles refer to the potential release of fragments or debris due to failures of rotating equipment, such as a turbo-compressor rotor or blade failure.

3. External hazard events: These are events originating from outside the system being analyzed, which can pose risks to the system's safety and functionality. Examples of external hazard events include seismic events (earthquakes), aircraft crashes, or external floods. These events may have the potential to impact the system and multiple failures that can lead to undesired consequences.

When conducting a full scope completed PRA, it is crucial to consider all these categories to comprehensively assess and analyze the potential initiating events and associated risks.

### 4.7. Licensing Basis Events

The plant response to an IE forms the event sequence analyzed in a full scope PRA. These event sequences form the basis from which LBEs are selected. At this early stage of the conceptual design, a PRA has not been completed. PRAs and event sequence frequency quantification have been completed in the past for various similar reactor concepts such as ALLEGRO GFR, CEA GFR2400, GA Modular High-Temperature Gas-Cooled Reactor (MHTGR), Japan's High-Temperature Engineering Test Reactor, China's High Temperature Gas-Cooled Test Reactor, DOE's Next Generation Nuclear Plant project, and United Kingdom gas-cooled reactors. Based on these past studies, proposed LBE sequences have been selected for initial consideration as shown in Table 1. This initial list completes task 1 of the TI-RIPB LBE selection process described earlier in Section 4.4. As the design development progresses through conceptual, preliminary, and final design, the PRA will also go through a similar development process and this initial list of LBEs will be revised as discussed in tasks 2 through 4 in Section 4.4.

**Table 1. Selected LBE Sequences**

| No. | Initiating Event | Event Sequence | Expected Frequency per reactor year | Expected End State |
|---|---|---|---|---|
| 1a | Power TCG shutdown | Reactor trip, continued decay heat removal via power conversion system (PCS) | $10^{-1}$/yr. | Safe shutdown, return to power when source of TCG shutdown corrected |

| No. | Initiating Event | Event Sequence | Expected Frequency per reactor year | Expected End State |
|---|---|---|---|---|
| 1b | Power TCG Shutdown due to control rod withdrawal | Reactor trip function fails (transient without scram condition), PCS and RHR ineffective | $10^{-6}$ to $10^{-8}$/yr | Core damage, potential fuel melt with reactor coolant system (RCS) pressurized, possible high pressure melt ejection (HPME) loads on containment. |
| 1c | Power TCG shutdown due to leak in Precooler or intercooler | Reactor trip, RHR in both active and passive mode fails | $10^{-6}$ to $10^{-8}$/yr. | Core and reactor vessel damage, potential fuel melt with RCS pressurized, possible high pressure melt ejection (HPME) loads on containment. |
| 2a | Loss of offsite power and connection to grid | Reactor trip, PCS cooling capability lost, RHR cooling in either passive or active mode | $10^{-1}$ to $10^{-3}$/yr. | Safe shutdown, return to power when offsite power is restored. |
| 2b | Loss of offsite power and connection to grid | Reactor trip, PCS cooling capability lost, emergency diesel generators (EDGs) fail, RHR cooling via passive mode | $10^{-3}$ to $10^{-6}$/yr. | Safe shutdown in station blackout, return to power when offsite power restored or EDG repaired |
| 2c | Loss of offsite power and connection to grid | Reactor trip, PCS cooling capability lost, EDGs fail, RHR passive mode fails | $10^{-6}$ to $10^{-8}$/yr. | Core and reactor vessel damage, potential fuel melt with RCS pressurized, evolves more slowly than 1c, possible HPME loads on containment |
| 3a | Small leak (<10 mm) in RCS Pressure Boundary | Reactor trip, PCS cooling capability lost, RHR cooling in active mode | $10^{-1}$ to $10^{-3}$/yr. | Safe shutdown, leakage of RCS coolant into containment, need to cleanup containment and repair leak, then restart |
| 3b | Small leak (<10 mm) in RCS Pressure Boundary | Reactor trip function fails (transient without scram condition), PCS and RHR ineffective | $10^{-5}$ to $10^{-8}$/yr. | Core damage with RCS depressurizing, may evolve more quickly than 1b, but no HPME concerns |

| No. | Initiating Event | Event Sequence | Expected Frequency per reactor year | Expected End State |
|---|---|---|---|---|
| 3c | Small leak (<10 mm) in RCS Pressure Boundary | Reactor trip, PCS cooling capability lost, RHR active cooling fails | $10^{-4}$ to $10^{-7}$/yr. | Core damage with RCS depressurizing, evolves more quickly than 2b, but no HPME concerns |
| 4a | Moderate leak (<100 mm) in RCS Pressure Boundary | Reactor trip, PCS cooling capability lost, RHR cooling in active mode | $10^{-3}$ to $10^{-5}$/yr. | Safe shutdown, leakage of RCS coolant into containment, need to cleanup containment and repair leak, then restart |
| 4b | Moderate leak (<100 mm) in RCS Pressure Boundary | Reactor trip, PCS cooling capability lost, RHR active cooling fails | $10^{-5}$ to $10^{-8}$/yr. | Core damage with RCS depressurized, evolves more quickly than 3c, but no HPME concerns |
| 5a | Large leak (>100 mm) in RCS Pressure Boundary | Reactor trip, PCS cooling capability lost, RHR cooling in passive mode | $10^{-5}$ to $10^{-7}$/yr. | Core damage with RCS depressurized, evolves more quickly than 3c, but no HPME concerns |
| 5b | Large leak (>100 mm) in RCS Pressure Boundary | Reactor trip, PCS cooling capability lost, RHR cooling in passive mode, containment breach | $10^{-6}$ to $10^{-8}$/yr. | Core damage with RCS depressurized, evolves more quickly than 3c, but no HPME concerns |
| 6a | Turbine missile | Reactor trip, PCS cooling capability lost, RHR cooling in either passive or active mode | $10^{-5}$ to $10^{-8}$/yr. | Safe shutdown, potential small leak in power conversion vessel |
| 7a | Design Basis Seismic Event (0.3 g PGA) | Reactor Trip, PCS fails, RCS pressure boundary intact, RHR cooling in active or passive mode | $10^{-4}$ to $10^{-5}$/yr. | Safe shutdown, no damage to any safety related SSCs |
| 7b | Severe Seismic Event (1-2 g) | Reactor Trip, PCS fails, RCS leaks or breaks, RHR active cooling fails, containment and passive cooling intact | $10^{-5}$ to $10^{-7}$/yr. | Core damage, potential fuel melt with RCS depressurized, containment intact |

| No. | Initiating Event | Event Sequence | Expected Frequency per reactor year | Expected End State |
|---|---|---|---|---|
| 7c | Design Basis Wind, Fire, or Flood Event | Reactor Trip, PCS fails, RCS pressure boundary intact, RHR cooling in active or passive mode | $10^{-4}$ to $10^{-5}$/yr. | Safe shutdown, no damage to any safety related SSCs |
| 7d | Severe Wind, Fire, or Flood Event | Reactor Trip, PCS fails, RCS pressure boundary intact, RHR active cooling fails, containment and passive cooling intact | $10^{-5}$ to $10^{-7}$/yr. | Safe shutdown, potential core damage, no release from containment |

## 5. SUMMARY AND CONCLUSIONS

The FMR LBE selection approach is consistent with the LMP as described in NEI 18-04 and associated reference documents. The LBE selection approach has the following attributes:

- Systematic and Reproducible
- Reasonably Complete
- Provides Timely Input to Design Decisions
- Risk-Informed and Performance-Based
- Consistent with Applicable Regulatory Requirements

The LBEs presented in Section 4 are based on historical precedents from gas-cooled reactor design and licensing efforts. At this early stage of the conceptual design, a PRA has not been started but insights from past PRA efforts have been applied. The approach is designed to ensure that an appropriate set of limiting events are reflected in the selection of DBAs and that the full set of LBEs define the risk significant events. This is essential to ensure that risk insights are appropriately reflected in the design and licensing decisions.

The events in Section 4 have not been explicitly categorized into AOOs, DBEs, BDBEs and DBAs at this time. Initial categorization will be done at the end of the conceptual design effort based on a process hazard analysis. Ultimately, event categorization will be risk-informed consistent with LMP and NEI 18-04 once a full-scope PRA is completed.




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