Review of LBE Selection for Domestic Applicability from TI-RIPB

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1. Introduction

To demonstrate the safety of nuclear power plants against potential failures of SSCs (Structures, Systems, and Components), human errors, and natural disasters, it is essential to conduct transient and accident analyses. These analyses are critical for evaluating and assessing the safety of nuclear power plants. The first step in this process involves selecting appropriate events and accident for the plant.

For large Light Water Reactors (LWRs), event and accident selection has traditionally been based on methodologies such as Failure Modes and Effects Analysis (FMEA) and Hazard and Operability Study (HAZOP) in the United States. These methods, combined with operational experience, have led to the establishment of a comprehensive list of events and accidents for large LWRs. When Korea imported large LWRs from the United States, it adopted these event and accident lists as a foundation.

The experience gained from event classification in large LWRs can be beneficial in selecting events and accidents for PWR-based i-SMRs and SMART 100. Standards and guidance such as ANSI/ANS-51.1, ANSI/ANS-58.14, or NRC Regulatory Guide 1.70, which are based on light water reactors, can be applied in these cases.

However, the recently developed Non-LWRs, such as Sodium-cooled Fast Reactors (SFRs), High-Temperature Gas-cooled Reactors (HTGRs), and Molten Salt Reactors (MSRs), face limitations when applying the event classification systems originally designed for light water reactors. These Non-LWRs are being developed simultaneously across a wide range of designs and sizes, each with unique characteristics. The lack of operational experience and the incomplete designs further complicate the application of traditional, reactor-type-specific event classification frameworks. Consequently, there is a need to develop a new event classification system capable of accommodating the diverse range of designs associated with these Non-LWRs.

1.1. Domestic Licensing Demand for Advanced Reactor

Globally, various countries and companies are actively engaged in the development of SMRs. The reactors currently under development include not only traditional PWRs but also a diverse range of designs such as HTGRs, SFRs, MSRs, and Heat-pipe reactors. According to a 2024 report by the OECD/NEA, a total of 69 SMRs are under development, with 71% of these reactors classified as Non-LWRs[1]. Additionally, these reactors are being designed for various applications, including hydrogen production, floating nuclear power plants, district heating, and electricity generation.

In Korea, the potential licensing demand for PWR type SMRs includes i-SMRs, SMART 100, and BANDI. In addition, reactors under development include Non-LWRs such as MSRs, HTGRs, and SFRs. Globally, the demand for the development and licensing of Advanced Reactors(AR) [1] is being driven primarily by private companies, a trend that is also evident in Korea. Korea Atomic Energy Research Institute (KAERI), for example, is to develop a nuclear propulsion ship based on MSR. Additionally, Denmark's Seaborg Technologies is planning to introduce an MSR-based floating nuclear power plant in Korea.

1.2. 10 CFR Part 53 Rulemaking in U.S.

The United States, a long-time leader in Non-LWR development, has already identified the challenges associated with licensing these ARs. In response, the U.S. has been researching regulatory approaches that can comprehensively address the diverse range of designs, sizes, and modular reactors. The NRC is developing a new licensing rule, 10 CFR Part 53, which is designed to encompass both LWR and Non-LWRs. The draft rule for 10 CFR Part 53 was issued in the second half of 2024, with the final rule expected to be published by July 2025[2].

The NRC proposes 10 CFR Part 53 as an optional, technology-inclusive regulatory framework. The regulatory requirements developed under this rule aim to provide flexible and practical evaluation methods for ARs, with a particular emphasis on using risk-informed and performance-based (RIPB) approaches. Although 10 CFR Part 53 offers two independent frameworks, Options A and B, this paper focuses exclusively on Framework A.

Framework A of 10 CFR Part 53 introduces a RIPB-based event classification methodology specifically designed for ARs especially Non-LWR. This paper provides a brief overview of the TI-RIPB methodology,

[1] An AR refers to a next generation nuclear reactor characterized by innovative design concepts that distinguish it significantly from traditional large LWRs, regardless of the plant size, installation type, or reactor type.

which serves as the foundation for 10 CFR Part 53 Framework A, and examines whether this event classification methodology, based on U.S. regulations, can be applied to domestic regulatory environments.

2. TI-RIPB Methodology

2.1 Introduction of TI-RIPB Methodology

Framework A of 10 CFR Part 53 was developed based on the outcomes of the Licensing Modernization Project (LMP), which was led by Southern Company in the late 2010s with support from the U.S. Department of Energy (DOE). The primary goal of the LMP was to reduce regulatory uncertainty in order to accelerate the commercialization of ARs. To achieve this, the LMP developed a systematic, risk-informed, performance-based, and predictable methodology known as the Technology-Inclusive, Risk-Informed, and Performance -Based (TI-RIPB) methodology. The LMP proposed an enhanced licensing framework based on this TI-RIPB methodology to the NRC, which, after review, documented it in the industry guidance NEI 18-04[3]. Subsequently, the NRC endorsed NEI 18-04 through Regulatory Guide 1.233[4], incorporating the TI-RIPB methodology as the foundation for Framework A of 10 CFR Part 53.

The TI-RIPB methodology is defined as follows:

- Technology-Inclusive (TI): This approach incorporates the technological characteristics of all reactor types, allowing the application of each reactor type's specific mechanical source terms.
- Risk-Informed (RI): It utilizes information obtained from systematic risk assessments and applies additional structured prescriptive rules to address uncertainties not covered by the risk assessment.
- Performance-Based (PB): It utilizes quantifiable performance metrics related to the frequency and outcomes of Licensing Basis Events and evaluates the effectiveness of SSCs based on their performance requirements in mitigating these events.

The regulatory process developed through the TI-RIPB methodology comprises three primary components: the selection of Licensing Basis Events (LBEs), the safety classification of SSCs, and the determination of defense-in-depth (DID) adequacy. The selection of LBEs provides a systematic definition, classification, and evaluation of events for nuclear power plants. The safety classification of SSCs offers a systematic approach to categorizing the safety significance SSCs associated with these LBEs. Finally, the adequacy of DID is assessed to ensure comprehensive evaluation of the identified LBEs and SSCs, ensuring that all safety measures are sufficient to manage risks effectively.

2.2 Elements of TI-RIPB Methodology

The TI-RIPB methodology is designed to encompass a wide range of reactor technologies, integrating both deterministic and probabilistic assessment results into a risk-informed approach. Additionally, it provides a performance-based framework that uses quantitative metrics to evaluate the risk significance of events[5].

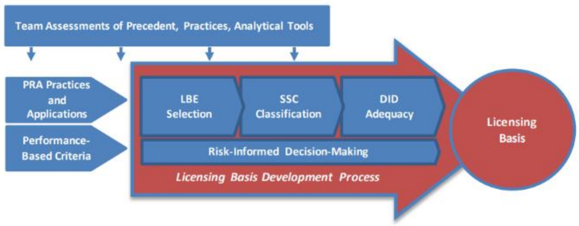

Figure 1 Elements of TI-RIPB Methodology

As discussed in Section 2.1, the elements of the TI-RIPB methodology, as shown in Figure 1, consist of three key components: LBE selection, safety classification of SSCs, and evaluation of defense-in-depth (DID) adequacy. To carry out these processes, probabilistic risk assessment (PRA) is conducted from the early stages of design, and its results are utilized throughout the process. The methodology also outlines the development of performance-based criteria for SSCs applied to the plant. For aspects that cannot be fully evaluated through PRA, the DID adequacy assessment enhances the reliability of the results.

The development process of the TI-RIPB licensing framework is not a one-time task but is instead completed through multiple iterative cycles. This process is repeated as needed during the conceptual design, basic design, and detailed design phases, thereby improving the reliability of the design.

3. Selection of LBEs

The first step of the TI-RIPB process, and the foundation of the new event classification methodology for Non-LWRs, is the selection of LBEs. These events are termed "Licensing Basis Events" because they serve as the foundational data for all licensing activities. The term LBE is an umbrella that encompasses the traditional event categories used in previous classification methods, including Anticipated Operational Occurrences (AOO), Design Basis Events (DBE), Design Basis Accidents (DBA), and Beyond Design Basis Events (BDBE).

3.1 LBEs Selection Process

The selection and evaluation process for LBEs is illustrated in Figure 2. The process begins with the identification of LBEs using PRA in the early stages. Subsequently, LBEs are categorized into AOO, DBE, and BDBE according to the event frequency defined in TI-RIPB (see Appendix). Each LBE is evaluated based on its frequency of occurrence and its radiological consequences as identified in the PRA results.

After the individual evaluation of each LBE, an integrated risk assessment for the entire plant is conducted. Finally, the process concludes with an evaluation of the adequacy of DID. A distinguishing feature of this process, compared to traditional event classification methods, is the mandatory use of PRA to identify and select LBEs. This integration of PRA into the selection process ensures a more comprehensive and risk-informed approach to event classification.

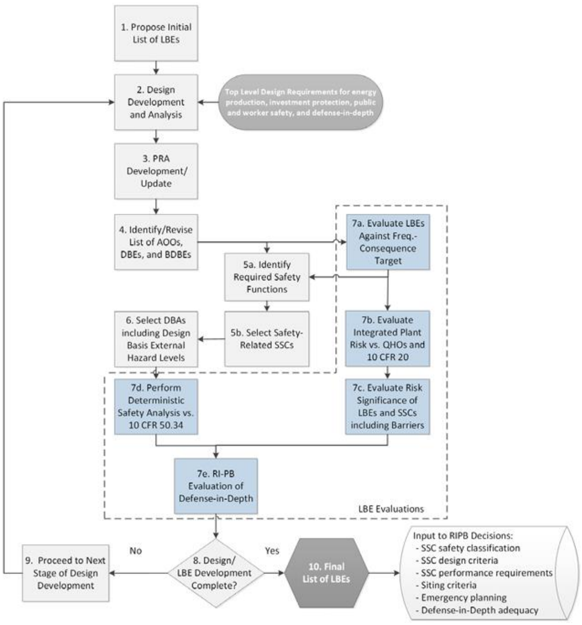

Figure 2 Process for Selecting and Evaluating LBEs

3.2 Criteria of F-C Target

The evaluation of LBEs is based on risk, specifically the product of event frequency and consequence, as illustrated in Figure 3. The vertical axis represents the frequency of events that could occur annually at the plant, with the unit being '#/plant-year' rather than '#/reactor-year' to account for multiple modular reactors. The horizontal axis indicates the cumulative Total Effective Dose Equivalent (TEDE) in rem over a 30-day period following an event.

In the TI-RIPB methodology, the guideline for LBE evaluation is established by combining the U.S. public dose limits with the event frequency criteria defined by TI-RIPB (see Appendix). This guideline is represented by the Frequency-Consequence (F-C) Target (blue line) in Figure 3. The TI-RIPB methodology recommends that all individual LBEs should be positioned to the left of this F-C Target.

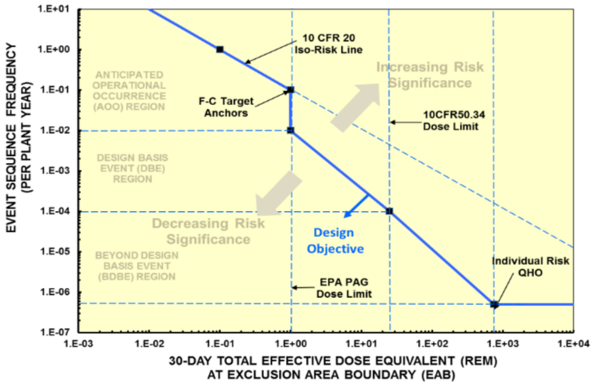

Figure 3 Frequency-Consequence Evaluation Criteria Proposed for TI-RIPB

The F-C Target is set based on the application of four specific U.S. dose criteria, which include:

- 10CFR20 iso-Risk Line
- EPA PAG Dose Limit
- 10CFR50.34 Dose Limit
- Individual Risk QHO

In the AOO region, which covers a frequency range of 1 to 10^{-2}/plant-year, the F-C Target is set by combining the dose limits from 10 CFR 20 and the EPA PAGs. According to 10 CFR 20, the annual TEDE for the public from plant operations must not exceed 0.1 rem (1 mSv). Consequently, the risk for all LBEs within this region must remain below this threshold. In Figure 3, the F-C Target line follows the 10 CFR 20 iso-Risk curve until it intersects with the EPA PAG Dose Limit, where the graph then bends downward. If an LBE's consequence exceeds 1 rem, offsite emergency response must be initiated in accordance with the EPA PAG Dose Limit. To avoid triggering offsite emergency response, the consequences of AOOs are limited to no more than 1 rem.

In the DBE region, which covers a frequency range of 10^{-2} to 10^{-4}/plant-year, there is no need to consider offsite emergency response, allowing for an expanded consequence criterion compared to AOOs. The upper consequence limit for DBEs is determined by the 10 CFR 50.34 dose limit. Under 10 CFR 50.34, in the event of a radiological release, individuals at the Exclusion Area Boundary (EAB) must not receive more than 25 rem TEDE.

In the BDBE region, which covers a frequency range of 10^{-4} to $5 \cdot 10^{-7}$/plant-year, the F-C Target's upper consequence and lower frequency limits are governed by the Quantitative Health Objectives (QHO) set forth in the NRC Safety Goal Policy. As a result, the BDBE region is defined by the area to the left of the line connecting the 25 rem at the 10^{-4}/plant-year to the 750 rem, which corresponds to the higher early fatality risk threshold in the QHO, at the $5 \cdot 10^{-7}$/plant-year.

3.3 Criteria of entire plant

In Section 3.2, the F-C Target criteria are addressed, which apply to each individual LBE. In addition to the guidelines for each LBE, the TI-RIPB methodology also establishes cumulative risk targets for the integrated set of LBEs. The TI-RIPB methodology defines three cumulative risk targets:

- The total mean frequency of exceeding a site boundary dose of 100 mrem from all LBEs should not exceed 1/plant-year. This metric is introduced to ensure that the consequences from the entire range of LBEs from higher frequency, lower consequences to lower frequency, higher consequences are considered. The value of 100 mrem is selected from the annual cumulative exposure limits in 10 CFR 20.
- The average individual risk of early fatality within 1 mile of the EAB from all LBEs based on mean estimates of frequencies and consequences shall not exceed 5×10-7 /plant-year to ensure that the NRC Safety Goal QHOs for early fatality risk is met.
- The average individual risk of latent cancer fatalities within 10 miles of the EAB from all LBEs based on mean estimates of frequencies and consequences shall not exceed 2×10-6 /plant-year to ensure that the NRC Safety Goal QHOs for latent cancer fatality risk is met.

4. Domestic Application of LBE Selection Methodology

4.1 Terminology Definition in TI-RIPB

The TI-RIPB methodology classifies LBEs based on event frequency (see Appendix). The AOO, DBE, BDBE, and DBA defined within the TI-RIPB methodology have been modified to align with the NRC's terminology, facilitating their application within the TI-RIPB framework without conflicting with NRC's regulatory definitions. The Appendix compares the event types as defined by the NRC and the TI-RIPB methodology. When comparing the event classification system in Regulatory Guide 1.70, which is applied in Korea, with that defined by the TI-RIPB methodology, it is observed that they share similar categories, as illustrated in Fig. 4. Therefore, it is anticipated that the frequency ranges for AOO, DBE, and BDBE could be applicable within the domestic context as well.

To comply with NRC Regulatory Guide 1.203, "Transient and Accident Analysis Methods," it is necessary to evaluate not only AOO, DBE, and BDBE but also DBA to demonstrate the design safety of nuclear power plants through safety analysis. The TI-RIPB methodology defines some events within the DBE and BDBE categories as deterministic DBA. According to the NRC's definition of DBA presented in Appendix, a DBA is characterized as postulated accident that a nuclear facility must be designed and built to withstand without loss to the SSCs necessary to ensure public health and safety. Similarly, the DBA defined by the TI-RIPB methodology consists of events that can be mitigated and prevented solely by safety-related SSCs, without the involvement of non-safety related SSCs.

As reviewed above, if the Korean nuclear regulatory body adopts Regulatory Guide 1.233, which endorses the TI-RIPB methodology, for domestic Non-LWR reactors, it is believed that deriving and evaluating LBEs based on this methodology, as well as applying them to design, are fully feasible.

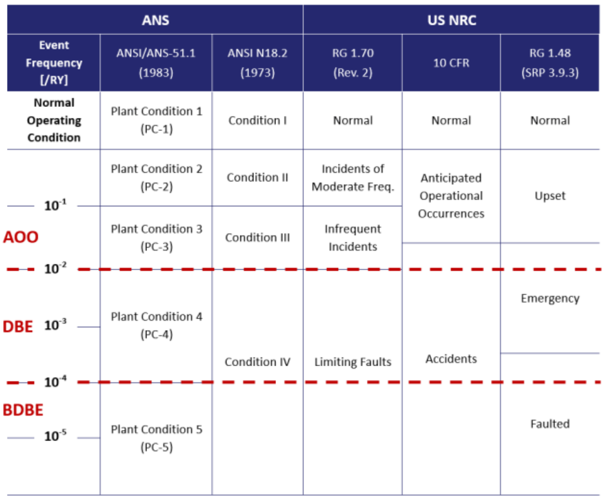

Figure 4 Comparison with the LWR Event Classification

4.2 Review of F-C Target and entire plant Criteria

The comparison between the dose limit criteria introduced by the TI-RIPB methodology for distinguishing each LBE and the corresponding U.S. and Korean regulations is presented in Table 1 below.

Table 1 F-C Target Criteria Regulation

TI-RIPB Criteria	Reference Criteria	U.S. Regulation	Korea Regulation
0.1 rem / plant-year	0.1 rem / reactor-year	10CFR20.1301, “Dose Limit to the Public”	Enforcement Decree of Nuclear Safety Act [Annex 1], “Dose Limits”
1 rem / 30 days	1 rem / 4 days	EPA PAG (EPA-400/R-17/001)	Enforcement Rules of the Act on Physical Protection and Radiological Emergency [Annex 4], "Criteria for offsite emergency response"

25 rem / 30 days	25 rem / the duration of the incident	10CFR50.34[2] and 10CFR100.11, "Determination of EAB, LPZ, and PDC distance"	NSSC Notices 2017-15, "Technical Standards for Location of Nuclear Reactor Facilities"
750 rem / 30 days	750 rem / at once	high probability of early fatality (ICRP 2), NRC Safety Goal QHOs for early fatality risk	-
$5 \cdot 10^{-7}$/ plant-yr	$5 \cdot 10^{-7}$/ plant-yr		

The first column of Table 1 presents the frequency and consequence criteria required by the F-C Target, while the second column shows the corresponding U.S. reference criteria. The dose limit specified by 10 CFR 20.1301 is based on a reactor-year basis; however, the F-C Target is conservatively set on a plant-year basis, considering that many Non-LWR reactors are modular. The remaining three consequence criteria are also based on the 30-day TEDE values, presenting slightly more stringent requirements compared to the existing reference criteria.

When comparing U.S. and Korean Regulation, the 10 CFR 20.1301 Dose Limit is equivalent to the "Dose Limit" criteria specified in the Enforcement Decree of the Nuclear Safety Act [Annex 1]. Similarly, the EPA PAG standard corresponds to the "Criteria for offsite emergency response" in the Enforcement Rules of the Act on Physical Protection and Radiological Emergency [Annex 4]. Additionally, NSSC Notice 2017-15, "Technical Standards for Location of Nuclear Reactor Facilities," directly references 10 CFR 100.11. However, the QHO from the U.S. NRC safety goals are not reflected in domestic regulations.

Table 2 Integrated Plant Risk Criteria Regulation

TI-RIPB Criteria	U.S. Regulation	Korea Regulation
The total mean frequency of exceeding a site boundary dose of 100 mrem from all LBEs should not exceed 1/plant-year.	10CFR20.1301, "Dose Limit to the Public"	Enforcement Decree of Nuclear Safety Act [Annex 1], "Dose Limits"
The average individual risk of early fatality within 1 mile of the EAB from all LBEs based on mean estimates of frequencies and consequences shall not exceed 5×10^-7 /plant-year.	NRC Safety Goal QHOs for early fatality risk	-
The average individual risk of latent cancer fatalities within 10 miles of the EAB from all LBEs based on mean estimates of frequencies and consequences shall not exceed 2×10^-6 /plant-year.	NRC Safety Goal QHOs for latent cancer fatality risk	-

Table 2 presents the dose evaluation criteria for the entire plant. The 100 mrem value is selected based on the annual cumulative exposure limit from 10 CFR 20, which aligns with the "Dose Limit" specified in the Enforcement Decree of the Nuclear Safety Act [Annex 1].

Additionally, it is proposed to meet the early fatality risk and latent cancer fatality risk criteria of the NRC Safety Goal QHO. NSSC Notice No. 2017-34, " Regulation on the Scope of Accident Management and the Detailed Criteria for Evaluating Accident Management Capabilities," Article 9 (Risk Evaluation) incorporates the NRC Safety Goal as the target value for PRA, 'Prompt and cancer fatalities among nearby residents due to the operation of and accidents at the nuclear power reactor facility shall satisfy 0.1% or lower of the sum of the total risk or the equivalent performance goal value'. However, the QHO, which are the safety goals of the NRC, are not reflected in any domestic laws.

4.3 Review of LBEs Selection process

In the LBEs Selection process, the role of PRA is critically important. Since this process applies to all Non-LWR nuclear power plants regardless of reactor type, it is essential to apply an agreed-upon standard for Non-LWR PRA. ASME/ANS developed a PRA standard specifically for Non-LWRs (ASME/ANS RA-S-1.4-2021), which has been endorsed by the NRC through the Trial Regulatory Guide 1.247. Currently, this Regulatory Guide is undergoing pilot application, and it is planned to be revised into an official Regulatory Guide after receiving feedback from U.S. nuclear stakeholders.

5. Conclusions

Upon reviewing the F-C Target Criteria for the Selection of LBEs methodology, it was determined that most of the criteria can be adequately replaced with existing domestic regulations. The 750 rem exposure QHO criterion in the TI-RIPB methodology is a hypothetical scenario with an extremely low probability, where everyone would die within 1 to 2 months. Therefore, we do not anticipate any significant changes from reflecting it in Korea.

This paper reviews the applicability of the Selection of LBEs, a top-tier component of the TI-RIPB methodology, within the domestic context. The remaining processes, namely SSC Classification and adequacy of DID, will be addressed in subsequent discussions. Further review and discussion are required to assess the domestic applicability of these two processes.

[2] 10 CFR 50.34 requires compliance with 10 CFR 100 regarding site characteristics. Moreover, the dose limits mandated by 10 CFR 50.34(a)(1)(ii)(D) are identical to those required by 10 CFR 100.11.

In conclusion, the three TI-RIPB methodologies - Selection of LBEs, SSC Classification, and adequacy of DID - are systematically and sequentially structured. These processes involve multiple iterations and back-fits throughout the plant design phase. As a result of these iterative processes, the plant ultimately achieves both safety and effectiveness, ensuring a comprehensive and robust design.

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Appendix. Definitions of LBEs

Event Type	NRC Definition	TI-RIPB Definition
Anticipated Operational Occurrences (AOOs)	"Conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit* and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power." [SRP 15.0 and 10 CFR 50 Appendix A]	Anticipated event sequences expected to occur one or more times during the life of a nuclear power plant, which may include one or more reactor modules. Event sequences with mean frequencies of 1×10-2/plant-year and greater are classified as AOOs. AOOs take into account the expected response of all SSCs within the plant, regardless of safety classification.
Design Basis Events (DBEs)	"Conditions of normal operation, including AOOs, design-basis accidents, external events, and natural phenomena, for which the plant must be designed to ensure functions of safety-related electric equipment that ensures the integrity of the reactor coolant pressure boundary; the capability to shut down the reactor and maintain it in a safe shutdown condition; or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures." [SRP 15.0]	Infrequent event sequences that are not expected to occur in the life of a nuclear power plant, which may include one or more reactor modules, but are less likely than AOOs. Event sequences with mean frequencies of 1×10-4/plant-year to 1×10-2/plant-year are classified as DBEs. DBEs take into account the expected response of all SSCs within the plant regardless of safety classification.
Beyond Design Basis Events (BDBEs)	"This term is used as a technical way to discuss accident sequences that are possible but were not fully considered in the design process because they were judged to be too unlikely. (In that sense, they are considered beyond the scope of design-basis accidents that a nuclear facility must be designed and built to withstand.) As the regulatory process strives to be as thorough as possible, 'beyond design-basis' accident sequences are analyzed to fully understand the capability of a design." [NRC Glossary]	Rare event sequences that are not expected to occur in the life of a nuclear power plant, which may include one or more reactor modules, but are less likely than a DBE. Event sequences with mean frequencies of 5×10-7/plant-year to 1×10-4/plant-year are classified as BDBEs. BDBEs take into account the expected response of all SSCs within the plant regardless of safety classification.
Design Basis Accidents (DBA)	"Postulated accidents that are used to set design criteria and limits for the design and sizing of safety-related systems and components." [SRP 15.0] "A postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety." [NRC Glossary and NUREG-2122]	Postulated event sequences that are used to set design criteria and performance objectives for the design of SR SSCs. DBAs are derived from DBEs based on the capabilities and reliabilities of SR SSCs needed to mitigate and prevent event sequences, respectively. DBAs are derived from the DBEs by prescriptively assuming that only SR SSCs are available to mitigate postulated event sequence consequences to within the 10 CFR 50.34 dose limits.
Licensing Basis Events (LBEs)	Term not used formally in NRC documents.	The entire collection of events considered in the design and licensing basis of the plant, which may include one or more reactor modules. LBEs include AOOs, DBEs, BDBEs, and DBAs.